Appendix A

Revision 0

APPENDIX A

Oconee Plant Specific Data

This Appendix contains the plant specific data and limits for the Oconee Nuclear Station. The thermal hydraulic statistical core design was performed as described in the main body of this report.

Plant Specific Data

This analysis is for the Oconee plant (two loop B&W PWR) with Mark-B fuel assemblies detailed in Reference 2. The BWC critical heat flux correlation described in Reference 9 is used.

Thermal Hydraulic Code and Model

The VIPRE-01 thermal hydraulic computer code (Reference 1) and the Oconee eight channel model approved in Reference 2 are used in this analysis.

Statepoints

The statepoint conditions evaluated in this analysis are listed in Table A-1.

Key Parameters and Uncertainties

The key parameters and their uncertainty magnitude and associated distribution used in this analysis are listed on Table A-2. The range of key parameter values is listed on Table A-4.

DNB Statistical Design Limits

The statistical design limit for each statepoint evaluated is listed on Table A-3. Section 1 of Table A-3 contains the 500 case runs and Section 2 contains the 3000 case runs. All statepoint SDL values reported in this analysis are normally distributed. The statistical design limit using the BWC CHF correlation for Oconee was determined to be [ ]

Figure A-1 graphically depicts the application [ ]

TABLE A-1. Oconee SCD Statepoints

TABLE A-1 Continued Oconee SCD Statepoints

NOTES:

- 100% design flow is equal to four times 88,000 gpm/pump or 352,000 gpm total system flow.
- 100% Full Power (FP) is equal to 2568 MWth.

(1) Outlet temperature equals 581.0 °F.

(2)

TABLE A-2. Oconee Statistically Treated Uncertainties

Parameter	Type of Uncertainty	Type of Distribution	Uncertainty	Standard Deviation
Reactor System				
Power	Measurement	Normal	± 2.0 %FP	± 1.0 %FP
Temperature	Measurement	Normal	± 2.0 °F	± 1.0 °F
Pressure	Measurement	Normal	± 30.0 psi	± 15.0 psi
Core Flow	Measurement	Normal	± 2.0 %design	± 1.0 %design
Nuclear				
FΔh	Calculation	Normal	-----------	± 2.84 %
F_z	Calculation	Normal	-----------	± 2.91 %
z	Calculation	Uniform	± 6.0 in.	----------
Fq*	Calculation	Normal	[	]
Fq	Calculation	Normal	[	]
Hot Channel Flow Area	Measurement	Uniform	[ ]	----------
DNBR	Correlation	Normal	-----------	± 8.88 %
DNBR	Code	Normal	[	]

TABLE A-2 Continued Oconee Statistically Treated Uncertainties

Parameter	Justification
System Pressure	This uncertainty accounts for random uncertainties in various instrumentation components. Since the random uncertainties are normally distributed, the square root of the sum of the squares (SRSS) that results in the pressure uncertainty is also normally distributed.
Inlet Temperature	Same approach as System Pressure uncertainty.
Core Power	The core power uncertainty was calculated by statistically combining the various random uncertainties associated with the measurement of core power. Since the random uncertainties are normally distributed, the srss of them that results in the core power uncertainty is also normally distributed.
Core Flow	Same approach as Core Power uncertainty.
Radial Power, $F\Delta h$	This uncertainty accounts for the error associated in the physics code's calculation of radial assembly power and the measurement of the assembly power. This uncertainty distribution is normal.
Axial Peak Power, Fz	This uncertainty accounts for the axial peak prediction uncertainty of the physics codes. The uncertainty is normally distributed.
Axial Peak Location, z	This uncertainty accounts for the possible error in interpolating on axial peak location in the maneuvering analysis. The uncertainty is one half of the physics code's axial node. The uncertainty distribution is conservatively applied as uniform.

TABLE A-2 Continued Oconee Statistically Treated Uncertainties

Parameter	Justification
Local Heat Flux HCF, Fq"	This uncertainty accounts for the decrease in DNBR at the point of MDNBR due to engineering tolerances. This uncertainty is also increased to account for flux depression at the spacer grids. The uncertainty is normally distributed and conservatively applied as one-sided in the analysis to ensure the MDNBR channel location is consistent for all cases.
Rod Power HCF, Fq	This uncertainty accounts for the increase in rod power due to manufacturing tolerances. The uncertainty in calculating the peak pin from assembly radial peak is also statistically combined with the manufacturing tolerance uncertainty to arrive at the correct value. The uncertainty is normally distributed and conservatively applied as one-sided in the analysis to ensure the MDNBR channel location is consistent for all cases.
Hot Channel Flow Area	This uncertainty accounts for manufacturing variations in the instrument guide tube sub-channel flow area. This uncertainty is uniformly distributed and is conservatively applied as one-sided in the analysis to ensure the MDNBR channel location is consistent for all cases.
DNBR - Correlation	This uncertainty accounts for the CHF correlation's ability to predict DNB. The uncertainty is normally distributed.
DNBR - Code/Model	This uncertainty accounts for the thermal-hydraulic code uncertainties and offsetting conservatism's. This uncertainty also accounts for the small DNB prediction differences between various model sizes. This uncertainty is normally distributed.

TABLE A-3. Oconee Statepoint Statistical Results

500 Case Runs

Statepoint #
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
17
18
19
20
21
22
23
24
25
26
27
28
29
30
31
32
33
34
35
36
37
38
39
40
41

TABLE A-3 Continued Oconee Statepoint Statistical Results

500 Case Runs

Statepoint #
42
43
44
45
46
47
48
49
50
52
53
54
55
56
57
58
59
60
61
62
63
64
65
66
67
68
69
70
71
72
73
74
75
76
77
78
79
80
81
82

TABLE A-3 continued Oconee Statepoint Statistical Results

3000 Case Runs

Statepoint #
2-T
3-T
6-T
20-T
24-T
26-T
29-T
34-T
39-T
41-T
44-T
53-T
54-T
59-T
62-T
63-T
68-T
72-T
78-T

TABLE A-4 Oconee Key Parameter Ranges

Parameter	Maximum	Minimum

All values listed in this table are based on the currently analyzed Statepoints. Ranges are subject to change based on future statepoint conditions.

FIGURE A-1

Appendix B

Revision 0

APPENDIX B

McGuire/Catawba Plant Specific Data

This Appendix contains the plant specific data and limits for the McGuire and Catawba Nuclear Stations. The thermal hydraulic statistical core design was performed as described in the main body of this report.

Plant Specific Data

This analysis is for the McGuire and Catawba plants (four loop Westinghouse PWR's) with either Mark-BW or Optimized Fuel Assemblies as described in Reference 3. The BWCMV critical heat flux correlation described in Reference 9 is used for analyzing both fuel types.

Thermal Hydraulic Code and Model

The VIPRE-01 thermal hydraulic computer code (Reference 1) and the McGuire/Catawba eight channel model approved in Reference 3 are used in this analysis.

Statepoints

The statepoint conditions evaluated in this analysis are listed in Table B-1.

Key Parameters and Uncertainties

The key parameters and their uncertainty magnitude and associated distribution used in this analysis are listed on Table B-2. The range of key parameter values is listed on Table B-4.

DNB Statistical Design Limits

The statistical design limit for each statepoint evaluated is listed on Table B-3. Section 1 of Table B-3 contains the 500 case runs and Section 2 contains the 3000 case runs. All statepoint SDL values listed in this analysis are normally distributed. The statistical design limit using the BWCMV CHF correlation for McGuire/Catawba was determined to be [ ] Figure B-1 graphically depicts the application [ ]

TABLE B-1. McGuire/Catawba SCD Statepoints

Stpt #	Power	Pressure	Temperature	Flow	Axial Peak	Radial Peak
1						
2						
3						
4						
5						
6						
7						
8						
9						
10						
11						
12						
13						
14						
15						
16						
17						
18						
19						
20						
21						
22						
23						

TABLE B-1 - Continued McGuire/Catawba SCD Statepoints

Stpt #	Power	Pressure	Temperature	Flow	Axial Peak	Radial Peak
24						
25						
26						
27						
28						
29						
30						
31						
32						
33						
34						
35						
36						
37						
38						
39						
40						
41						
42						
43						

TABLE B-2. McGuire/Catawba Statistically Treated Uncertainties

Parameter	Uncertainty / Standard Deviation	Type of Distribution
Core Power	+/- 2% / +/- 1.22%	Normal
Core Flow		
Measurement	+/- 2.2% / +/- 1.34%	Normal
Bypass Flow	+/- 1.5%	Uniform
Pressure	+/- 30 psi	Uniform
Temperature	+/- 4 deg F	Uniform
$F^N_{\Delta H}$		
Measurement	+/- 3.25% / 1.98%	Normal
$F^E_{\Delta H}$	+/- 3.0% / 1.82%	Normal
Spacing	+/- 2.0% / 1.22%	Normal
F_Z	+/- 4.41% / 2.68%	Normal
Z	+/- 6 inches	Uniform
DNBR		
Correlation	+/- 16.78% / 10.2%	Normal
Code/Model	[ ]	Normal

TABLE B-2 Continued McGuire/Catawba Statistically Treated Uncertainties

Parameter	Justification
Core Power	The core power uncertainty was calculated by statistically combining the uncertainties of the process indication and control channels. The uncertainty is calculated from normally distributed random error terms such as sensor calibration accuracy, rack drift, sensor drift, etc combined by the square root sum of squares method (SRSS). Since the uncertainty is calculated from normally distributed values, the parameter distribution is also normal.
Core Flow Measurement	Same approach as Core Power.
Bypass Flow	The core bypass flow is the parallel core flow paths in the reactor vessel (guide thimble cooling flow, head cooling flow, fuel assembly/baffle gap leakage, and hot leg outlet nozzle gap leakage) and is dependent on the driving pressure drop. Parameterizations of the key factors that control ΔP, dimensions, loss coefficient correlations, and the effect of the uncertainty in the driving ΔP on the flow rate in each flow path, was performed. The dimensional tolerance changes were combined with the SRSS method and the loss coefficient and driving ΔP uncertainties were conservatively added to obtain the combined uncertainty. This uncertainty was conservatively applied with a uniform distribution.
Pressure	The pressure uncertainty was calculated by statistically combining the uncertainties of the process indication and control channels. The uncertainty is calculated from random error terms such as sensor calibration accuracy, rack drift, sensor drift, etc combined by the square root sum of squares method. The uncertainty distribution was conservatively applied as uniform.
Temperature	Same approach as Pressure.

TABLE B-2 Continued McGuire/Catawba Statistically Treated Uncertainties

Parameter	Justification
$F^N_{\Delta H}$ Measurement	This uncertainty is the measurement uncertainty for the movable incore instruments. A measurement uncertainty can arise from instrumentation drift or reproducibility error, integration and location error, error associated with the burnup history of the core, and the error associated with the conversion of instrument readings to rod power. The uncertainty distribution is normal.
$F^E_{\Delta H}$	This uncertainty accounts for the manufacturing variations in the variables affecting the heat generation rate along the flow channel. This conservatively accounts for possible variations in the pellet diameter, density, and U_{235} enrichment. This uncertainty distribution is normal and was conservatively applied as one-sided in the analysis to ensure the MDNBR channel location was consistent for all cases.
Spacing	This uncertainty accounts for the effect on peaking of reduced hot channel flow area and spacing between assemblies. The power peaking gradient becomes steeper across the assembly due to reduced flow area and spacing.. This uncertainty distribution is normal and was conservatively applied as one-sided to ensure consistent MDNBR channel location.
F_Z	This uncertainty accounts for the axial peak prediction uncertainty of the physics codes. The uncertainty distribution is applied as normal.
Z	This uncertainty accounts for the possible error in interpolating on axial peak location in the maneuvering analysis. The uncertainty is one half of the physics code's axial node. The uncertainty distribution is conservatively applied as uniform.
DNBR Correlation	This uncertainty accounts for the CHF correlation's ability to predict DNB. The uncertainty distribution is applied as normal.

TABLE B-2 Continued McGuire/Catawba Statistically Treated Uncertainties

Parameter		Justification
DNBR		
	Code/Model	This uncertainty accounts for the thermal-hydraulic code uncertainties and offsetting conservatisms. This uncertainty also accounts for the small DNB prediction differences between the various model sizes. The uncertainty distribution is applied as normal.

TABLE B-3. McGuire/Catawba Statepoint Statistical Results

500 Case Runs

Statepoint #
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
17
18
19
20
21
22
23
24
25
26
27
28
29
30
31
32
33
34
35
36
37
38
39
40
41
42
43

TABLE B-3 Continued McGuire/Catawba Statepoint Statistical Results

3000 Case Runs

Statepoint #	
2-T	
3-T	
4-T	
12-T	
13-T	
14-T	
16-T	
20-T	
37-T	
38-T	
39-T	

TABLE B-4 McGuire/Catawba Key Parameter Ranges

Parameter	Maximum	Minimum

All values listed in this table are based on the currently analyzed Statepoints. Ranges are subject to change based on future statepoint conditions.

FIGURE B-1

Appendix C and Response To Request for Additional Information

Revision 1

DPC-NE-2005-A

Duke Power Company Thermal-Hydraulic
Statistical Core Design Methodology

APPENDIX C

McGuire/Catawba Plant Specific Data

Mark-BW Fuel

BWU-Z CHF Correlation

Submitted April 1996

Approved November 1996

This Appendix contains the plant specific data and limits for the McGuire and Catawba Nuclear Stations with Mark-BW fuel using the BWU-Z form of the BWU critical heat flux correlation. The thermal hydraulic statistical core design analysis was performed as described in the main body of this report.

Plant Specific Data

This analysis is for the McGuire and Catawba plants (four loop Westinghouse PWR's) with Mark-BW fuel assemblies as described in Reference C-1. The parameter uncertainties and statepoint ranges were selected to bound the unit and cycle specific values of the McGuire and Catawba stations.

Thermal Hydraulic Code and Model

The VIPRE-01 thermal-hydraulic computer code described in Reference C-3 and the McGuire/Catawba eight channel code model approved in Reference C-1 are used in this analysis.

Critical Heat Flux Correlation

The BWU-Z form of the BWU critical heat flux correlation described in Reference C-2 is used for all statepoint analyses. This correlation was developed by BWFC for application to the Mark-BW fuel design. Reference C-2 was performed with the LYNXT thermal-hydraulic computer codes. The correlation was programmed into the VIPRE-01 thermal-hydraulic computer code by Duke Power Company and the BWU-Z CHF data base analyzed in its entirety. The results of this analysis are shown in Table C-1. The resulting Average M/P value and data standard deviation are within 1% of the values reported in Reference C-2.

Figures C-1 through C-5 graphically show the results of this evaluation. Figure C-1 shows there is no bias of measured CHF values to VIPRE-01 predicted values for the data base. Figure C-2 shows a histogram of the VIPRE-01 M/P ratios for the 530 point data base. Figures C-3 through C-5 show there is no bias with the VIPRE-01 calculated M/P ratios with respect to mass velocity, pressure, or thermodynamic quality. These figures compare closely with the same parameter representations in Reference C-2.

Based on the results shown in Table C-1 and Figures C-1 through C-5, the BWU-Z form of the BWU CHF correlation licensed in Reference C-2 can be used in DNBR calculations with VIPRE-01 for Mark-BW fuel.

Statepoints

The statepoint conditions evaluated in this analysis are listed in Table C-2. These statepoints represent the range of conditions to which the statistical DNB analyses limit will be applied.

Key Parameters and Uncertainties

The key parameters and their uncertainty magnitude and associated distribution used in this analysis are listed on Table C-2. The uncertainties were selected to bound the values calculated for each parameter at McGuire and Catawba. The resulting range of key parameter values generated in this analyses is listed on Table C-5.

DNB Statistical Design Limits

The statistical design limit for each statepoint evaluated is listed on Table C-4. Section 1 of Table C-4 contains the 500 case runs and Section 2 contains the 5000 case runs. The number of cases was increased from 3000 to 5000 as described in Attachment 1 of the main body of the report. All statepoint SDL values listed in this analysis are normally distributed. The maximum statepoint statistical DNBR value in Table C-4 for the 5000 case propagations was [ ].

Therefore, the statistical design limit using the BWU-Z form of the BWU CHF correlation for Mark-BW fuel at McGuire/Catawba was conservatively determined to be [ ].

FIGURE C-1

Measured CHF Versus Predicted CHF

Mark-BW Data Base

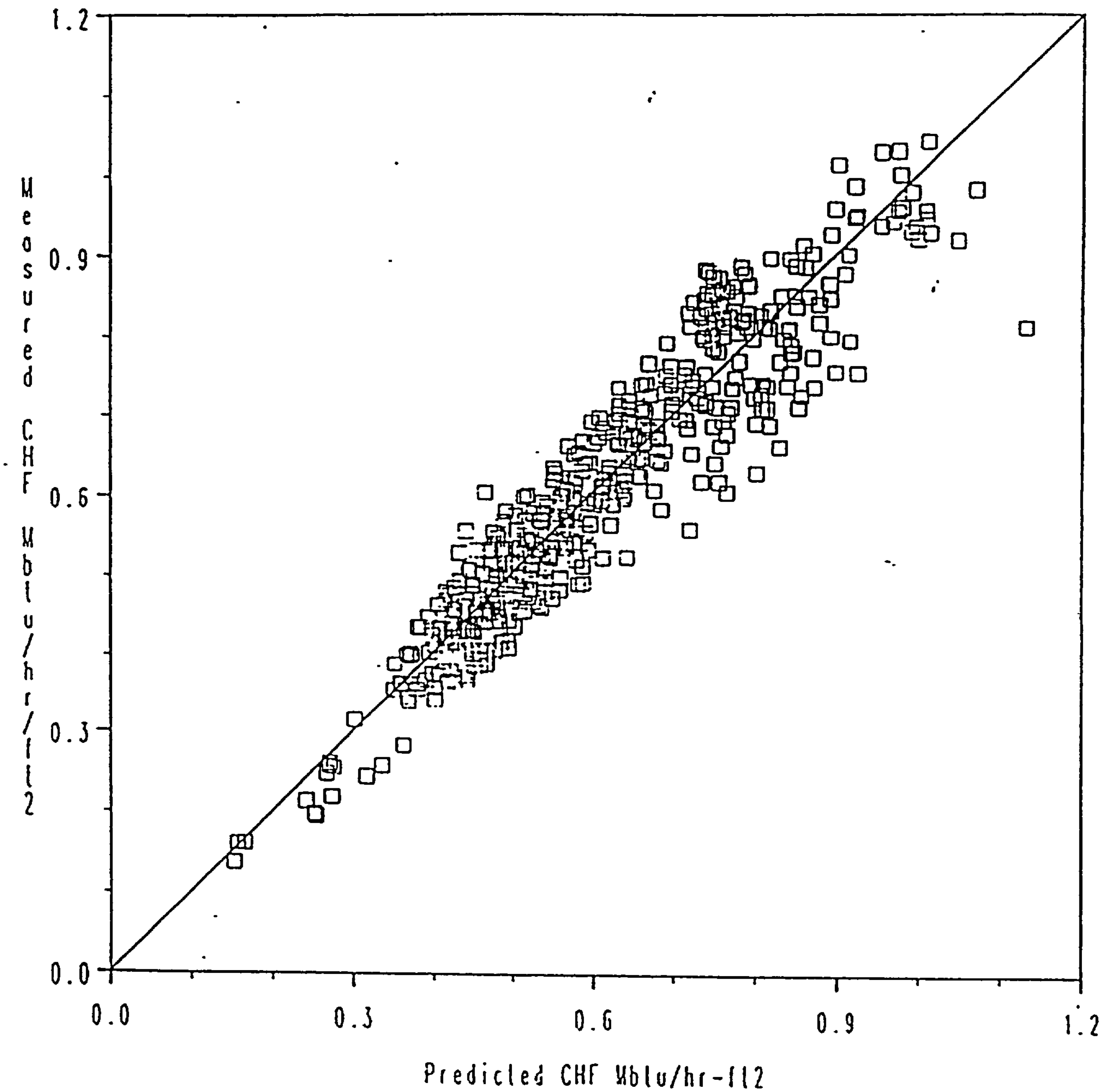

FIGURE C-2

Distribution of CHF Ratios

Mark-BW Data Base

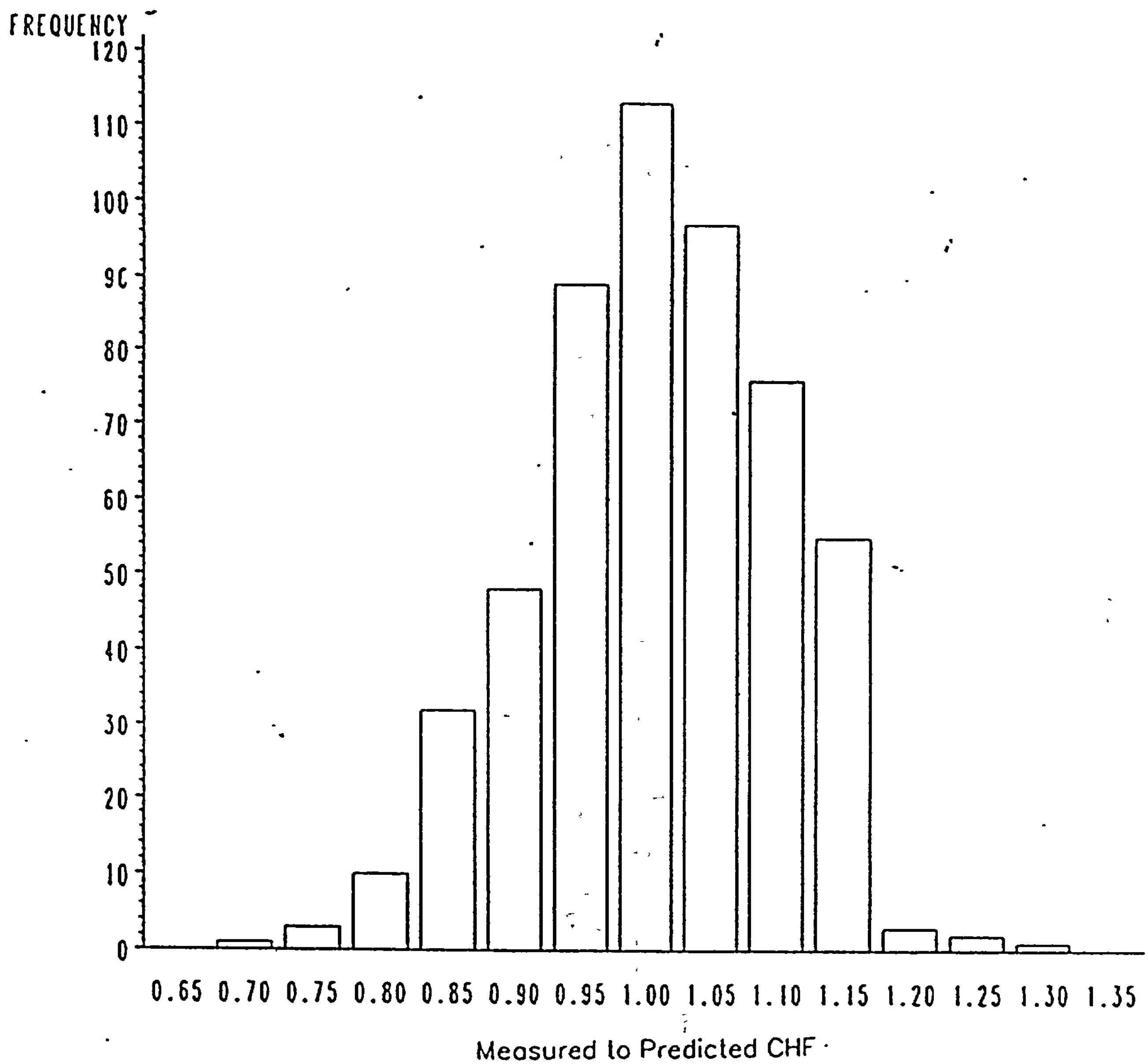

FIGURE C-3

Measured to Predicted CHF Versus Mass Velocity

Mark-BW Data Base

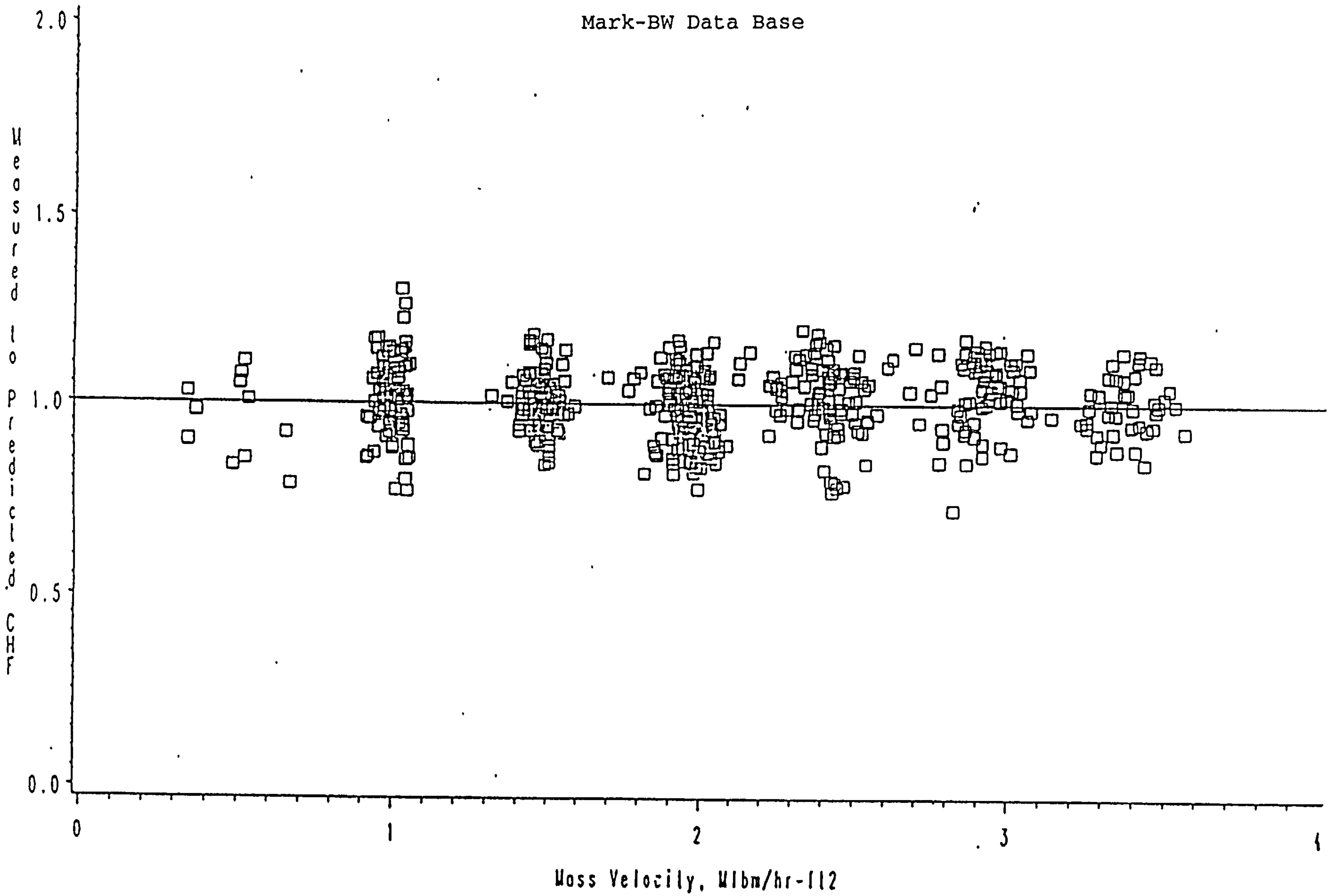

FIGURE C-4

Measured to Predicted CHF Versus Pressure

Mark-BW Data Base

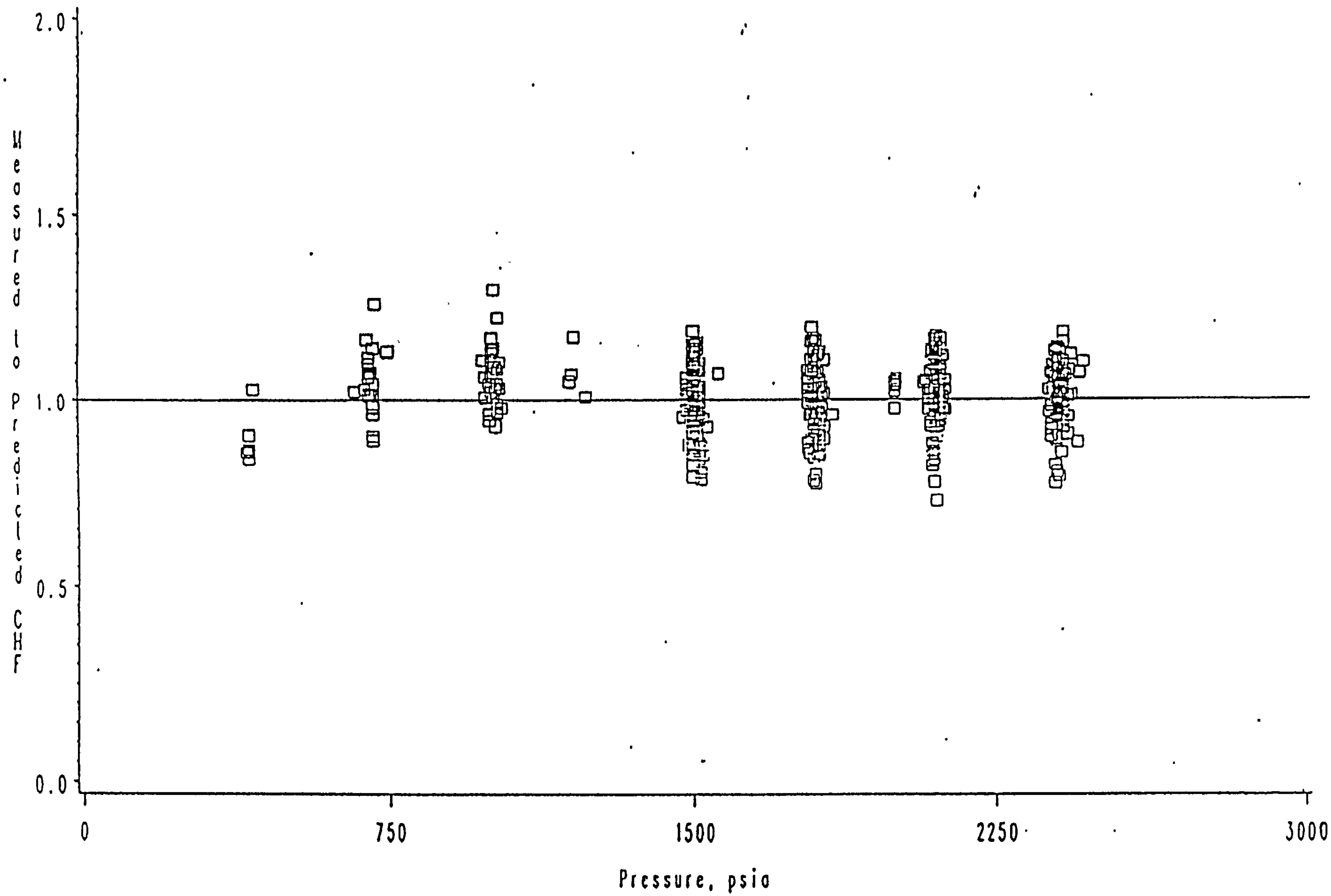

FIGURE C-5

Measured to Predicted CHF Versus Quality

Mark-BW Data Base

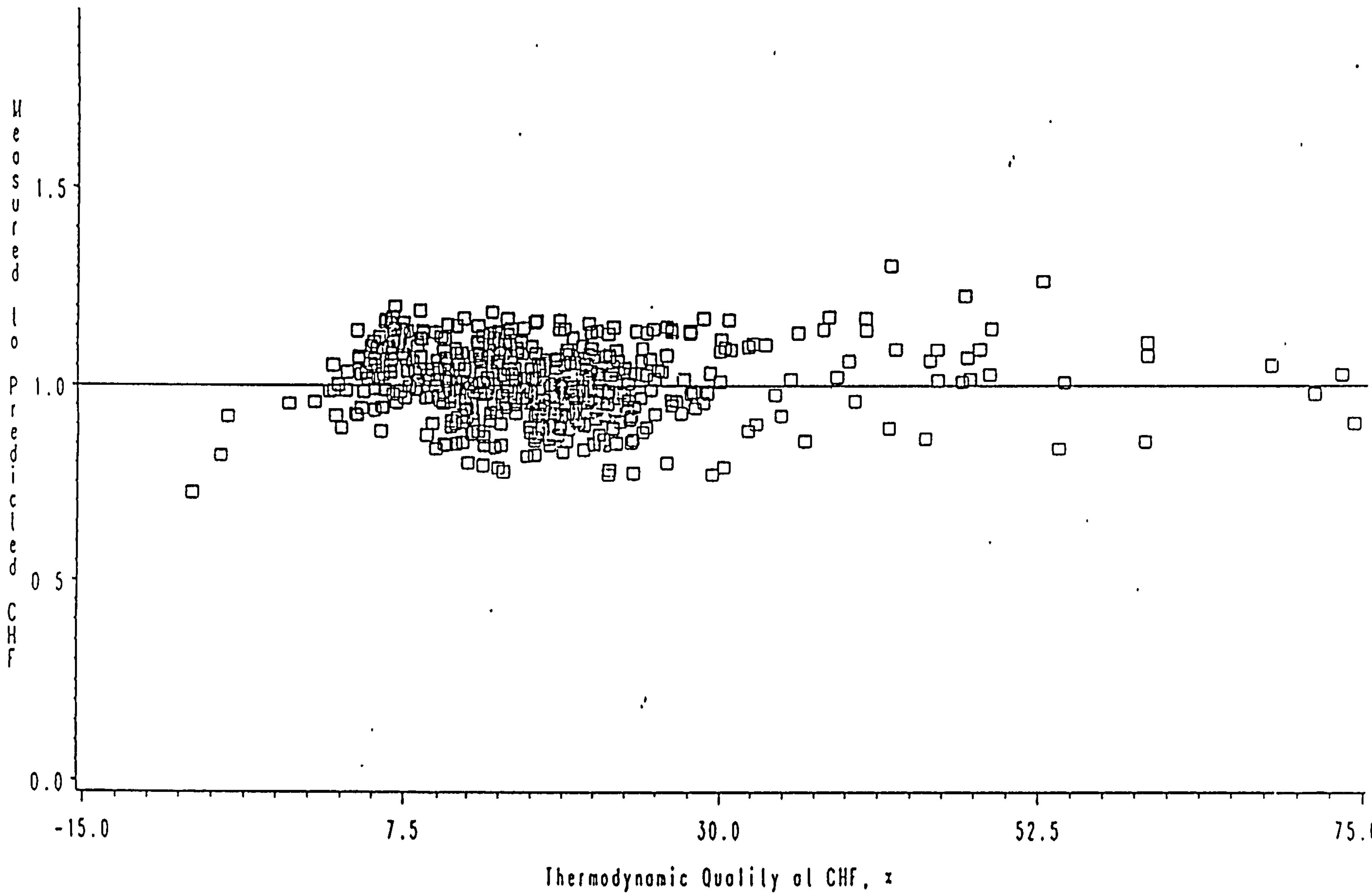

TABLE C-1 VIPRE-01 BWU-Z Correlation Verification

CHF Test Database Analysis Results

VIPRE-01 Statistical Results

Number Of Data Points	530
Average M/P	1.00850
Standard Deviation	0.09217
Upper D Prime	3469.0
Lower D Prime	3407.0
D Prime Value	3453.68
Accept Normality at 5% Level	

Parameter Ranges

Pressure, psia	400 to 2465
Mass Velocity, Mlbm/hr-ft^2	0.36 to 3.55
Thermodynamic Quality at CHF	less than 0.74
Thermal-Hydraulic Computer Code	VIPRE-01
Spacer Grid	Mark-BW 17x17
Design Limit DNBR, VIPRE-01	1.18

TABLE C-2 McGuire/Catawba SCD Statepoints

Stpt No.	Power* (% RTP)	RCS Flow (K gpm)	Pressure (psia)	Core Inlet Temperature (°F)	Axial Peak (F_z @ Z)	Radial Peak (FΔH)
1						
2						
3						
4						
5						
6						
7						
8						
9						
10						
11						
12						
13						
14						
15						
16						
17						
18						
19						
20						
21						
22						
23						
24						

* 100% RTP = 3411 Megawatts Thermal

TABLE C-3 McGuire/Catawba Statistically Treated Uncertainties

Parameter	Uncertainty / Standard Deviation	Type of Distribution
Core Power*	+/- 2% / +/- 1.22%	Normal
Core Flow		
Measurement	+/- 2.2% / +/- 1.34%	Normal
Bypass Flow	+/- 1.5%	Uniform
Pressure	+/- 30 psi	Uniform
Temperature	+/- 4 deg F	Uniform
$F^N_{\Delta H}$		
Measurement	+/- 4.0% / 2.43%	Normal
$F^E_{\Delta H}$	+/- 3.0% / 1.82%	Normal
Spacing	+/- 2.0% / 1.22%	Normal
F_Z	+/- 4.41% / 2.68%	Normal
Z	+/- 6 inches	Uniform
DNBR		
Correlation	+/- 15.25% / 9.27%	Normal
Code/Model	[ ]	Normal

* - Percentage of 100% RTP (68.22 MWth wherever applied).

TABLE C-3 Continued McGuire/Catawba Statistically Treated Uncertainties

Parameter	Justification
Core Power	The core power uncertainty was calculated by statistically combining the uncertainties of the process indication and control channels. The uncertainty is calculated from normally distributed random error terms such as sensor calibration accuracy, rack drift, sensor drift, etc. combined by the square root sum of squares method (SRSS). Since the uncertainty is calculated from normally distributed values, the parameter distribution is also normal.
Core Flow	
Measurement	Same approach as Core Power.
Bypass Flow	The core bypass flow is the parallel core flow paths in the reactor vessel (guide thimble cooling flow, head cooling flow, fuel assembly/baffle gap leakage, and hot leg outlet nozzle gap leakage) and is dependent on the driving pressure drop. Parameterizations of the key factors that control ΔP, dimensions, loss coefficient correlations, and the effect of the uncertainty in the driving ΔP on the flow rate in each flow path, was performed. The dimensional tolerance changes were combined with the SRSS method and the loss coefficient and driving ΔP uncertainties were conservatively added to obtain the combined uncertainty. This uncertainty was conservatively applied with a uniform distribution.
Pressure	The pressure uncertainty was calculated by statistically combining the uncertainties of the process indication and control channels. The uncertainty is calculated from random error terms such as sensor calibration accuracy, rack drift, sensor drift, etc. combined by the square root sum of squares method. The uncertainty distribution was conservatively applied as uniform.
Temperature	Same approach as Pressure.

TABLE C-3 Continued McGuire/Catawba Statistically Treated Uncertainties

Parameter	Justification
$F^N_{\Delta H}$	
Measurement	This uncertainty is the measurement uncertainty for the movable incore instruments. A measurement uncertainty can arise from instrumentation drift or reproducibility error, integration and location error, error associated with the burnup history of the core, and the error associated with the conversion of instrument readings to rod power. The uncertainty distribution is normal.
$F^E_{\Delta H}$	This uncertainty accounts for the manufacturing variations in the variables affecting the heat generation rate along the flow channel. This conservatively accounts for possible variations in the pellet diameter, density, and U_{235} enrichment. This uncertainty distribution is normal and was conservatively applied as one-sided in the analysis to ensure the MDNBR channel location was consistent for all cases.
Spacing	This uncertainty accounts for the effect on peaking of reduced hot channel flow area and spacing between assemblies. The power peaking gradient becomes steeper across the assembly due to reduced flow area and spacing. This uncertainty distribution is normal and was conservatively applied as one-sided to ensure consistent MDNBR channel location.
F_Z	This uncertainty accounts for the axial peak prediction uncertainty of the physics codes. The uncertainty distribution is applied as normal.
Z	This uncertainty accounts for the possible error in interpolating on axial peak location in the maneuvering analysis. The uncertainty is one half of the physics code's axial node. The uncertainty distribution is conservatively applied as uniform.

TABLE C-3 Continued McGuire/Catawba Statistically Treated Uncertainties

Parameter	Justification
DNBR	
Correlation	This uncertainty accounts for the CHF correlation's ability to predict DNB. The uncertainty distribution is applied as normal.
Code/Model	This uncertainty accounts for the thermal-hydraulic code uncertainties and offsetting conservatisms. This uncertainty also accounts for the small DNB prediction differences between the various model sizes. The uncertainty distribution is applied as normal.

TABLE C-4 McGuire/Catawba Statepoint Statistical Results

BWU-Z Critical Heat Flux Correlation

500 Case Runs

Statepoint #	Mean	σ	Coefficient of Variation	Statistical DNBR
1				
2				
3				
4				
5				
6				
7				
8				
9				
10				
11				
12				
13				
14				
15				
16				
17				
18				
19				
20				
21				
22				
23				
24				

TABLE C-4 Continued McGuire/Catawba Statepoint Statistical Results

BWU-Z Critical Heat Flux Correlation

5000 Case Runs

Statepoint #	Mean	σ	Coefficient of Variation	Statistical DNBR
1				
7				
9				
12				

TABLE C-5 McGuire/Catawba Key Parameter Ranges

Parameter	Maximum	Minimum
Core Power (% RTP)		
Pressure (psia)		
T inlet (deg. F)		
RCS Flow (Thousand GPM)		
FΔH, Fz, Z		

All values listed in this table are based on the currently analyzed Statepoints. Ranges are subject to change based on future statepoint conditions.

REFERENCES

C-1. DPC-NE-2004P-A, McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology Using VIPRE-01, December 1991.

C-2. The BWU Critical Heat Flux Correlations, BAW-10199-P, Babcock and Wilcox, Lynchburg, Virginia, December 1994 (SER received April 5, 1996).

C-3. VIPRE-01: A Thermal-Hydraulic Code For Reactor Cores, EPRI NP-2511-CCM-A, Vol. 1-4, Battelle Pacific Northwest Laboratories, August 1989.

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DUKE POWER

September 5, 1996

U. S. Nuclear Regulatory Commission
Washington, D. D. 20555

Attention: Document Control Desk

Subject: McGuire Nuclear Station
Docket Numbers 50-369 and -370
Catawba Nuclear Station
Docket Numbers 50-413 and -414
Use of BWU-Z Correlation by Duke Power;
Supplemental Information

By letter dated April 26, 1996, Duke Power requested NRC approval for use of the BWU-Z correlation at its McGuire and Catawba nuclear stations. A supplement was provided by letter dated December 4, 1995. The December 4, 1996 letter (paragraph 4) stated that the better thermal performance of the fuel can be used to reduce cycle fuel costs. This is due to fact the licensed BWU-Z correlation conservatively quantifies the inherent thermal margin of the Mark-BW17 fuel. This margin can be used in fuel cycle analyses to raise peaking, thereby saving fuel costs. Additionally, the December 4, 1996 letter contained a typographical error in the last sentence of Paragraph 5. The references identified should be 5 and 6, not 6 and 7 as the letter stated.

During telcons on August 21 and 27, 1996, between the NRC staff and Duke, additional information/clarification was requested by the Staff. Attached are the questions and associated responses.

Note that upon approval of the new Appendix C (to topical report DPC-NE-2005), which was transmitted by the April 26, 1996 letter and contains the technical basis for the use of BWU-Z, the topical report will be republished, including the new Appendix C, as DPC-NE-2005, Revision 1.

If there are any questions or additional information is required, please call Scott Gewehr at (704) 382-7581.

M. S. Tuckman

M. S. Tuckman

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Request for Additional Information To Topical Report DPC-NE-2005P, Appendix C

The questions are shown in italics and the responses immediately follow.

1) *What fuel type and core configuration are currently operating at McGuire and Catawba?*

McGuire and Catawba are both operating with a full (homogenous) core of Mark-BW17 fuel assemblies, also called Mark-BW 17x17. This will be the fuel type until a transition, beginning in the year 2000, to Westinghouse 17x17 mixing vane fuel. Transition to Westinghouse fuel will require licensing of a different critical heat flux correlation and corresponding statistical design limit applicable to that fuel type.

2) *Table C-1 lists the statistical results of the CHF test data base analysis with the VIPRE-01 thermal-hydraulic computer code. Explain the differences in values between this table and the table in BAW-10199P-A which documents the same data analysis with the LYNXT or LYNX 2 code.*

The information provided for the Mark-BW17 data base using the BWU-Z correlation on the top of page 4-3 in BAW-10199P-A shows the average M/P, Standard Deviation (corrected for N), and Design Limit DNBR (denoted DNBR(L)) for the test data when analyzed with LYNXT or LYNX 2. Table C-1 of the DPC-NE-2005 Appendix C is a direct comparison of the same analysis and the same test data with VIPRE-01 code. The VIPRE-01 code has a slightly higher average M/P and slightly lower standard deviation for the entire test data base when compared to LYNXT or LYNX 2. The combination of these two parameters gives the VIPRE-01 code a slightly lower Design Limit DNBR for the test data base.

The more conservative value for the parameter is selected by Duke Power Company (DPC) when performing an analyses. For example, the standard deviation listed in Table C-3 (DPC-NE-2005 Appendix C) for the correlation uncertainty is the higher of the LYNX and VIPRE-01 values rounded to two significant figures. The Design Limit DNBR calculated with VIPRE-01 is presented in Table C-1 for comparison only. The standard deviation is the only value that impacts the SCD calculation. If the BWU-Z form of the BWU correlation is used by DPC in non-SCD analyses, the larger of the two non-statistical correlation limits (the LYNX value listed on page 4-3 of BAW-10199P-A) will be used.

3) Explain the method used to calculated the 500 and 5000 case statistical DNBR values for each statepoint and how the statistical limit is used.

The method used to evaluated the BWU-Z form of the BWU correlation in Appendix C is identical to the procedure outlined in the main body of the DPC-NE-2005 report. This procedure is outlined starting in Section 2.0 on page 5. The key parameters listed in Section 2.1, page 6, are identical in Appendix C. The statepoints in Table C-2 of Appendix C were selected to bound the range of key parameters where the SCD analyses with BWU-Z will be applied.

The selection of uncertainties is discussed in Section 2.2, page 7, of DPC-NE-2005. Table C-3 lists the values used in the BWU-Z SCD analyses. These are identical to the values used in the BWCMV analysis (Appendix B of DPC-NE-2005) except for the correlation standard deviation (as explained in Question 2 above) and the FΔH measurement uncertainty which was increased slightly for the BWU-Z analysis.

The method for statepoint propagation is explained in Section 2.3, page 8, of DPC-NE-2005. The calculation of the statepoint statistical limit is explained in Section 2.4, page 9 through 11, of DPC-NE-2005. The equation for the SDL calculation is shown on page 10. Included in the equation are Chi Square and K factor multipliers to ensure a conservative limit based on the number of cases calculated. The mean and standard deviation values for a statepoint fluctuate slightly as the number of cases increase. Increasing the number of cases gives higher confidence that the data analyzed defines bounding behavior, therefore the multipliers are reduced. This ensures the SDL limit is equally conservative even though the final statistical DNBR value is smaller as the number of cases gets larger. An example of the way the values change with an increasing number of cases is shown on Table 1.

The main body of the report lists the number of cases as either 500 or 3000 per statepoint. The propagation method is identical regardless of the number of cases generated. In the response to Question 8 of Attachment II, Request For Additional Information, in DPC-NE-2005, DPC stated that the number of cases may be increased. This number was increased to 5000 for the BWU-Z analysis in Appendix C. As explained in the response to Question 8, this increase is consistent with the methodology and does not in any way reduce the conservatism of the SDL limit calculated.

The 5000 case number was selected as a balance between computer resources required for the calculations and the reduction in statistical uncertainty. For example, increasing the number of cases by two thirds from 3,000 to 5,000 reduces the K factor by 0.011 (from 1.692 at 3,000 to 1.681 at 5,000). Further increasing the number of cases to 10,000 would require another doubling of resources for the same K factor reduction (from 1.681 at 5000 to 1.670 at 10,000).

The 500 and 5000 case results for the BWU-Z analysis are listed in Table C-4 of Appendix C. As described in DPC-NE-2005, the 5000 case statepoints are selected based on the results of the 500 case statepoint propagations. The 5000 case runs are used to determine a conservative Statistical Design Limit (SDL) for the correlation SCD analyses. A value larger than the largest 5000 case statepoint statistical DNBR value is listed on page C-4. This is the statistical design limit that will be used in analyses with the BWU-Z form of the BWU correlation for Mark-BW fuel at McGuire and Catawba. The statistical design limit listed on page C-4 will be applicable to an analysis as long as all statepoint parameters fall between the Maximum and Minimum ranges listed on Table C-5.

TABLE 1

Statepoint 1 Values

Number Of Cases	Coefficient Of Variation*	Chi Square Multiplier	K Factor Multiplier	Statistical DNBR**
500	0.1514	1.05549	1.763	1.392
1000	0.1541	1.03848	1.727	1.382
1500	0.1528	1.03115	1.712	1.369
2000	0.1537	1.02684	1.703	1.368
2500	0.1534	1.02393	1.698	1.364
3000	0.1539	1.02179	1.692	1.363
3500	0.1538	1.02013	1.689	1.362
4000	0.1543	1.01880	1.686	1.361
4500	0.1540	1.01771	1.683	1.358
5000	0.1539	1.01681	1.681	1.357

* **Coefficient of Variation = Standard Deviation / Mean for the number of cases**

$$\textbf{**Statistical DNBR} = \frac{1}{[\,1 - \{(\text{Coeff. of Variation}) * (\text{Chi Square Mult.}) * (\text{K Factor Mult.})\}]}$$

Appendix D and Response To Request for Additional Information

Revision 2

DPC-NE-2005A

Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology

APPENDIX D

Oconee Plant Specific Data

Mark-B11 Fuel

Application of BWU-Z CHF Correlation to Mark-B11 Mixing Vane Spacer Grid Fuel Design

Submitted: April 1997
Approved: June 1999

This Appendix contains the plant specific data and limits for the Oconee Nuclear Station with Mark-B11 fuel using the BWU-Z form of the BWU critical heat flux correlation. The thermal hydraulic statistical core design analysis was performed as described in the main body of this report.

Plant Specific Data

This analysis is for the Oconee plant (two loop Babcock and Wilcox PWR's) as described in Reference D-1. The parameter uncertainties and statepoint ranges were selected to bound the unit and cycle specific values of the Oconee station. This analysis models the improved, small diameter, mixing vane grid, Mark-B fuel assembly denoted as the Mark-B11 design. Four lead test assemblies began operation in Oconee 2 in May of 1996. FCF is scheduled to issue a Mechanical Design Topical Report to the NRC in December 1997.

Thermal Hydraulic Code and Model

The VIPRE-01 thermal-hydraulic computer code described in Reference D-3 and the Oconee eight and nine channel models approved in Reference D-1 are used in this analysis. Due to the fuel assembly design change, some specific data supplementary to Table 3-1 in Reference D-1 requires updating. This data is listed in Table D-1. Table D-1 includes fuel rod, control rod, and instrument guide tube diameters, the number of mixing and non-mixing vane grids, and the fuel rod length. The following section compares Mark-B design fuel assemblies with the Mk-B11 fuel assemblies.

Previous Mark-B design fuel assemblies consisted of 0.430 inch diameter fuel rods with 2 inconel and 6 intermediate non-mixing vane zircaloy grids. The Mark-B11 fuel assembly design is composed of fuel pins with a 0.416 inch outside diameter and two inconel grids and six intermediate zircaloy grids, one non-mixing grid and five mixing vane grids. The higher pressure drop and higher cladding surface heat flux of the Mark-B11 design is offset by the larger flow area and the presence of the mixing vane grids to result in improved assembly thermal performance.

The VIPRE-01 models approved in Reference D-1 are used to analyze the Mark-B11 fuel with the following exceptions:

1) The Mark-B11 fuel assembly geometry information is listed in Table D-1.

2) The turbulent mixing factor has been changed from 0.01 to 0.038 for the Mark-B11 fuel assembly design due to the presence of

mixing vane grids. The numerical value was determined and provided by the fuel supplier. This is consistent with FCF's 17x17 Mark-BW fuel assembly product and has been confirmed by Mark-B11 LDV test data.

3) The bulk void fraction model was changed from the Zuber-Findlay model to the EPRI. The Zuber-Findlay bulk void model is applicable only to qualities below approximately 0.7 and is discontinuous at a quality equal to 1.0 (Reference D-3). The EPRI bulk void model is essentially the same as the Zuber-Findlay bulk void model except for the equation used to calculate the drift velocity (Reference D-3). This eliminates the discontinuity at a quality equal to 1.0. Therefore, the EPRI model provides a full range (i.e., void fraction range, 0 - 1.0) of applicability required for performing DNB calculations. Also, for overall model compatibility, the subcooled void model was changed from LEVY, as specified in Reference D-1, to the EPRI correlation for the Mark-B11 fuel.

To evaluate the impact of changing bulk void models on DNB prediction, forty-four Mark-B11 CHF test data points (Reference D-2) were compared using both the Levy/Zuber-Findlay and EPRI/EPRI subcooled void/bulk void combinations in VIPRE-01. These data points cover a pressure range of 1005 to 2425 psia and an inlet temperature range 361.3 to 604.3°F. The mass flux at the MDNBR location varied from 0.542 to 2.963 Mlbm/hr-ft^2. The void fraction at the MDNBR location varied from 0.106 to 0.711.

The equilibrium quality at the MDNBR location varied from -0.104 to 0.198. The results of this comparison are as follows:

	Levy/Zuber-Findlay	EPRI/EPRI
Minimum DNBR (Avg)	0.991	0.996

The minimum DNBR results show a minimal difference of 0.54% (0.005 in DNB). Therefore, the EPRI bulk void model and EPRI subcooled void correlation will be used in Mark-B11 analysis.

Critical Heat Flux Correlation

The NRC approved BWU-Z form of the BWU critical heat flux correlation with the Mark B11V multiplier described in Reference D-2 is used for all Mark-B11 analyses. This correlation was developed by FCF for application to the Mark-B11 fuel design. The analysis in Reference D-2 was performed with the LYNXT thermal-hydraulic computer codes. This correlation was programmed into the VIPRE-01 thermal-hydraulic computer code by Duke Power Company and the Mark B11V data base analyzed in its entirety. The results of this analysis are shown in Table D-2. The resulting Average M/P value, data standard deviation, and CHF correlation limit are within 1% of the values reported in Reference D-2, page E-4 (also shown on Table D-2 under LYNXT column).

Figures D-1 through D-4 graphically show the results of this evaluation. Figure D-1 shows there is no bias of measured CHF values to VIPRE-01 predicted values for the data base. Figures D-2 through D-4 show there is no bias with the VIPRE-01 calculated M/P ratios with respect to mass velocity, pressure, or thermodynamic quality. These figures compare closely with the same parameter representations in Reference D-2.

Based on the results shown in Table D-2 and Figures D-1 through D-4, the BWU-Z form of the BWU CHF application correlation with the Mark-B11V multiplier, licensed in Reference D-2, can be used in DNBR calculations with VIPRE-01 for Mark-B11 fuel.

Statistical Core Design Analysis

Statepoints

The statepoint conditions evaluated in this analysis are listed in Table D-3. These statepoints represent the range of conditions to which the statistical DNB analyses limit will be applied. The range of key parameter values analyzed is listed on Table D-6.

Key Parameters and Uncertainties

The key parameters and their uncertainty magnitude and associated distribution used in this analysis are listed on Table D-4. The uncertainties were selected to bound the values calculated for each parameter at Oconee. The uncertainties have not changed except for the rod power hot channel factor (Fq), core flow measurement, and DNBR correlation. The uncertainty for Fq has changed due to fuel design changes. The core flow measurement uncertainty was increased to ensure that it is bounding. This results in a more conservative SDL. The DNBR correlation uncertainty is the same as that stated in Reference D-2, page 4-3.

DNB Statistical Design Limits

The statistical DNBR limit for each statepoint evaluated is listed on Table D-5. Section 1 of Table D-5 contains the 500 case runs and Section 2 contains the 5000 case runs. The number of cases was increased from 3000 to 5000 as described in Attachment 1 of the main body of this report (DPC-NE-2005) and Appendix C (Reference D-4). All of the DNBR distributions are normally distributed. The maximum statistical DNBR value in Table D-5 (full core of Mark-B11 fuel) for 5000 propagations is [ ]. Therefore, the statistical design limit, using the BWU-Z form of the BWU CHF correlation with the Mark-B11V multiplier for Mark-B11 fuel at Oconee, is [ ] for the range of parameters given in Table D-6.

Transition Cores

The transition core model determines the impact of the geometric and hydraulic differences between the resident Mark-B10 series fuel and the new Mark-B11 design. The 9 channel model described in Reference D-1 is used to evaluate the impact of transition cores containing Mark-B11 fuel. In Figure 4-5 in Reference D-1, Mark-B11 fuel is used instead of Mark-B6/7 and Mark-B10F/G fuel instead of Mark-B5. Therefore, channels 1 - 7 are modeled as Mark-B11 fuel, Channel 8 is modeled as Mark-B10F/G fuel, and Channel 9 is modeled as Mark-B11 fuel. The transition core analysis models each fuel type in those respective locations with the correct geometry. The form loss coefficients for each fuel design are input so the effect of crossflow out of the higher pressure drop mixing vane grid (Mark-B11) fuel is calculated.

A transition core penalty is evaluated by determining the DNBR impact on a Mark-B11 limiting assembly when analyzed with the 9 channel model. Once determined, several methods are available to conservatively compensate for the penalty. One method of compensating for the reduction in DNB performance due to the hydraulic effects of the conservatively modeled transition core is to explicitly apply a penalty to the Mark-B11 fuel generic peaking limits based on a full Mark-B11 core. Another option is to calculate maximum allowable peaking limits specifically modeling the transition core loading pattern in the detailed 64 channel model approved in Reference D-1. These methods will be used, as necessary, to determine the DNB effect of transition cores.

To evaluate the statistical DNB impact of the transition core, the most limiting statistical DNB statepoint (Statepoint 22 on

Table D-5) was evaluated using the 9 channel model. This statepoint is designated TR22 in Table D-5. At 5000 cases, the statistical DNBR for statepoint TR22 is slightly greater than the limit for statepoint 22, but less than the statistical design limit, [ ]. Therefore, the statistical design limit, [ ], is bounding for Mark-B10/B11 transition cores; as well as, full Mark-B11 cores.

Figure D-1 - Mark-B11 Vane Data
VIPRE-01 Measured Versus Predicted CHF

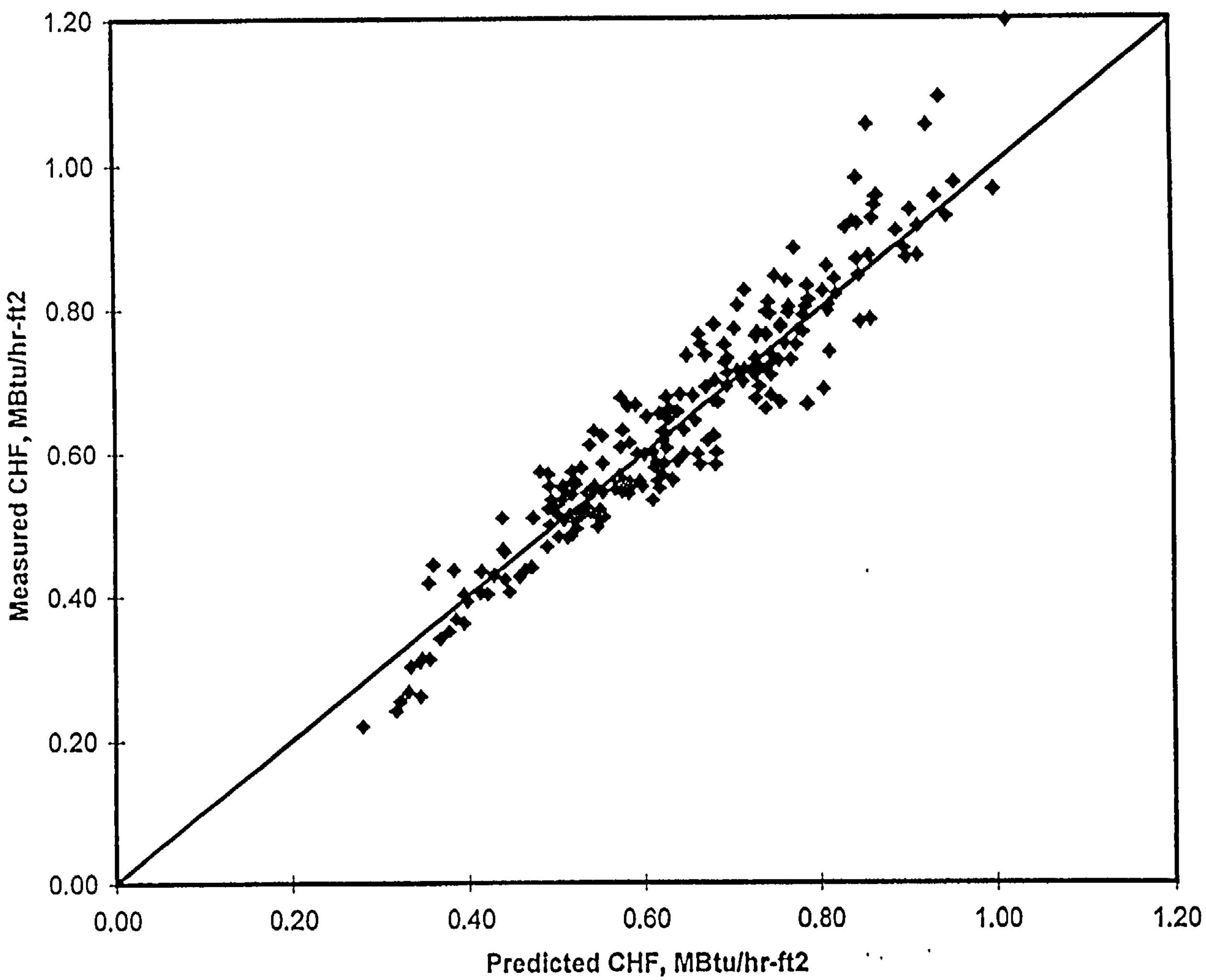

Figure D-2 - Mark-B11 Vane Data
VIPRE-01 Measured to Predicted CHF versus Mass Velocity

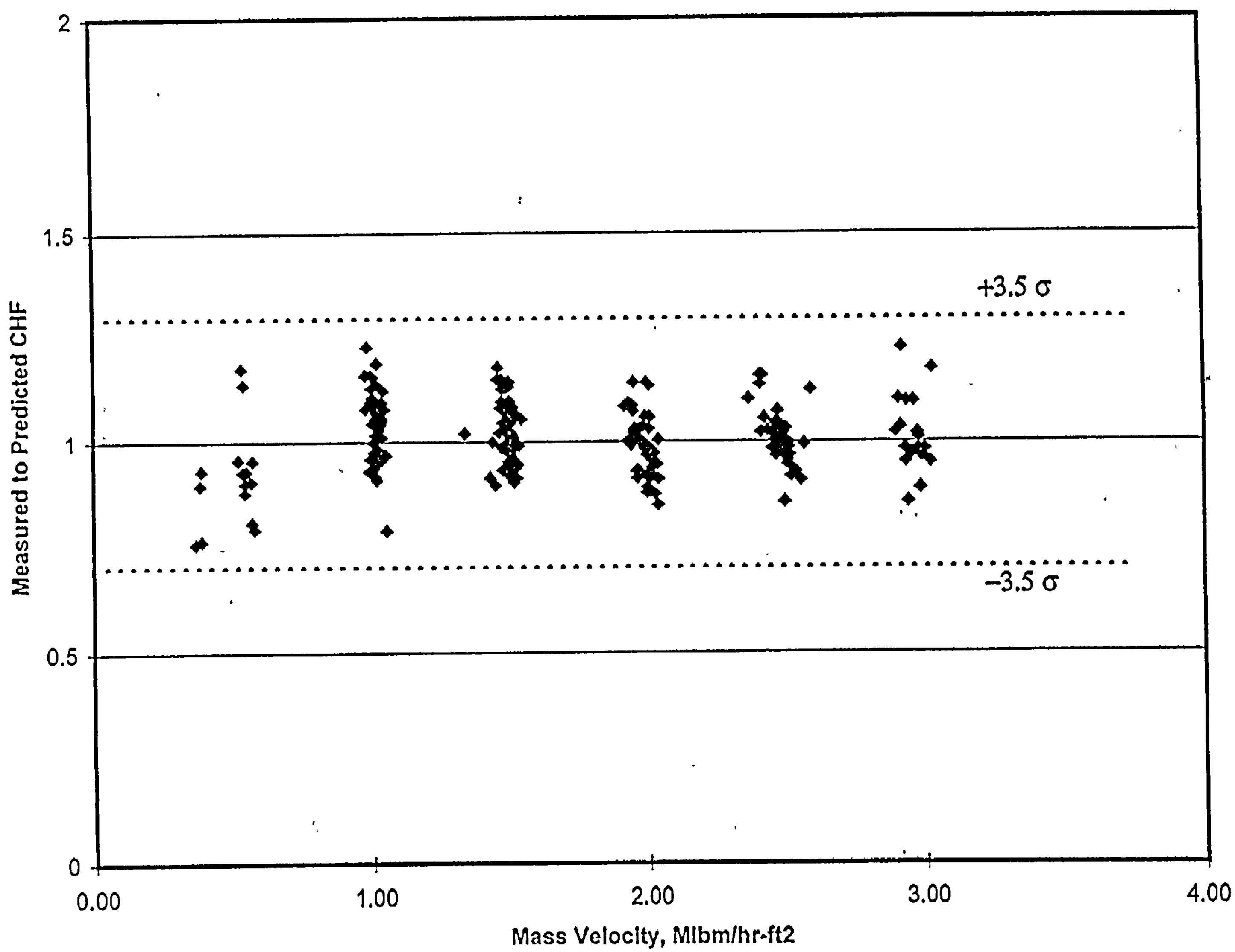

Figure D-3 - Mark-B11 Vane Data
VIPRE-01 Measured to Predicted CHF versus Pressure

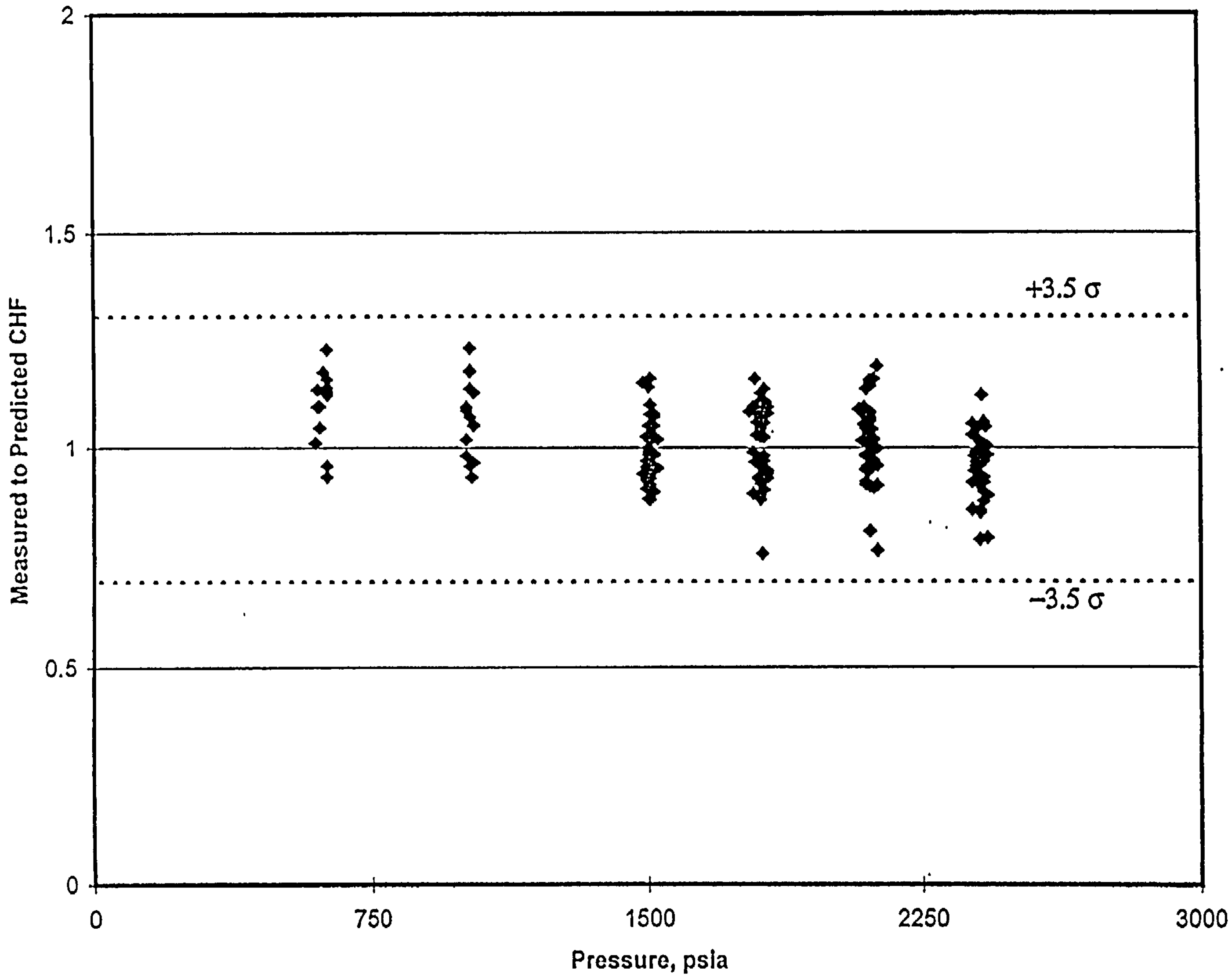

Figure D-4 - Mark-B11 Vane Data
VIPRE-01 Measured to Predicted CHF versus Quality

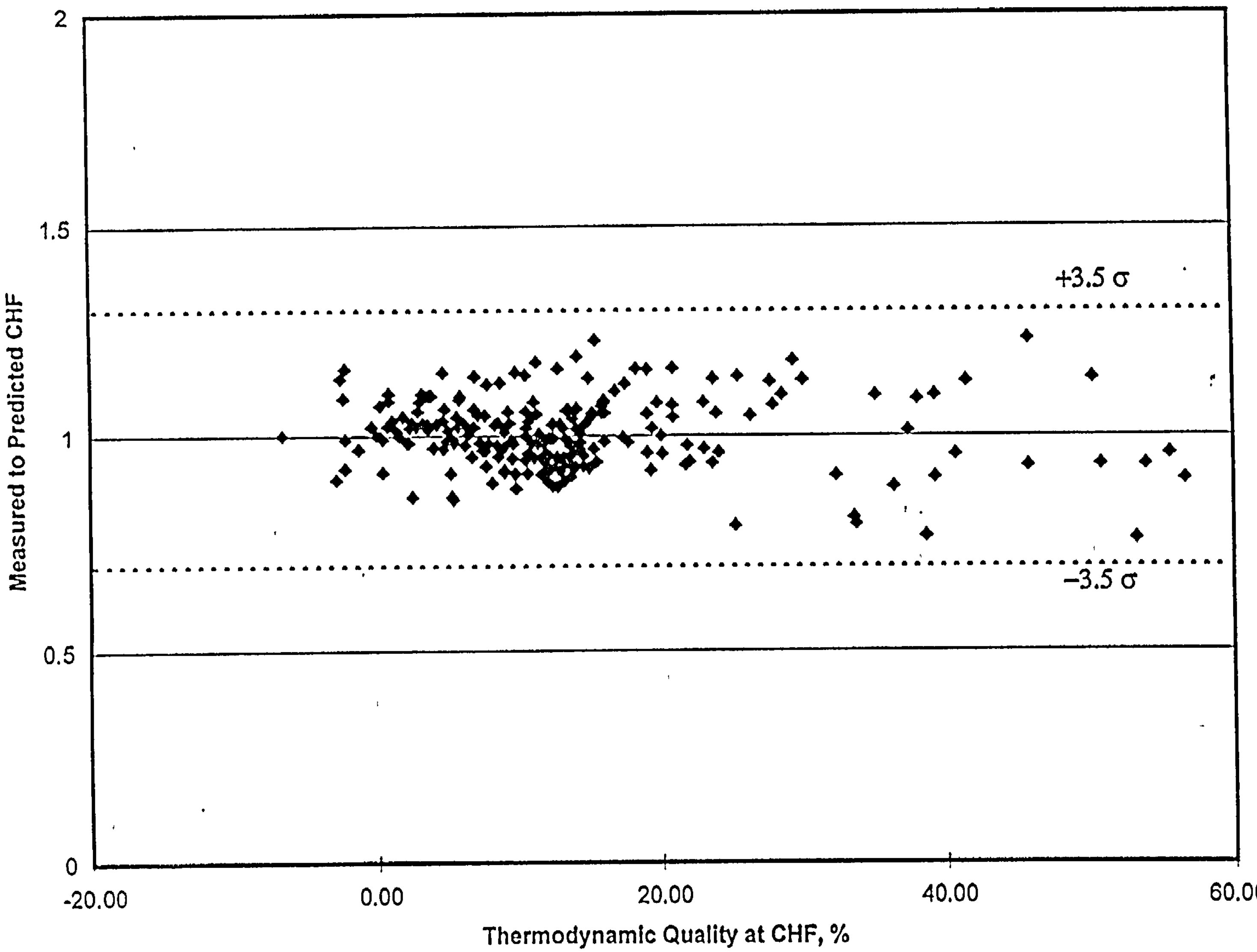

TABLE D-1 MARK-B11 FUEL ASSEMBLY DATA

(TYPICAL)

GENERAL FUEL SPECIFICATIONS

Fuel rod diameter, in. (Nom.)	0.416
Thimble tube diameter, in. (Nom.)	0.530
Instrument guide tube diameter, in. (Nom.)	0.554(1)/0.567(2)
Fuel rod pitch, in (Nom.)	0.568
Fuel assembly pitch, in. (Nom.)	8.587
Fuel rod length, in. (Nom.)	154.16

(1) Above lowest mixing vane grid (MV) and between MV grids.

(2) Below the first mixing vane grid and above the top of the last mixing vane.

GENERAL FUEL CHARACTERISTICS

Grids:	Material	Quantity	Location	Type
	Inconel	2	Upper and Lower	Non-Mixing Vane
	Zircaloy	6	Intermediate	1 Non-Mixing Vane, 5 Mixing Vane

Fuel Rods:	Material	Quantity
	Zircaloy-4	208

Fuel Cycle Design Assembly Features

Fuel Assy.	Mark
Designation:	B11
Features: vane	Smaller clad outside diameter and mixing grids.

TABLE D-2 VIPRE-01 BWU-Z Correlation with Mark-B11V Multiplier Verification

CHF Test Database Analysis Results

VIPRE-01/LYNXT Statistical Results

	VIPRE-01	LYNXT
n, # Of data	216	216
N, degrees of freedom (n-1)	215	215
M/P, Average measured to predicted CHF	1.0084	1.0040
σ (M/P/N)	0.0859	0.0868
K(215,0.95,0.95),one sided tolerance factor Ref. D-2)	1.830	1.830
DNBR(L)= 1/(M/P - Kσ)= 1/[1.0040 - 1.830(0.0868)]	1.175	1.183

Parameter Ranges

Pressure, psia	400 to 2465
Mass Velocity, Mlbm/hr-ft^2	0.36 to 3.55
Thermodynamic Quality at CHF	less than 0.74
Thermal-Hydraulic Computer Code	VIPRE-01
Spacer Grid	Mark-B11 15x15 Mixing Vane
Design Limit DNBR, VIPRE-01	1.19*

* The correlation design limit DNBR (1.19) applies only at or above a nominal pressure of 1000 psia (Reference D-2). In the low pressure region (below a nominal pressure of 1000 psia) the design limit DNBR in the following table will be used (Reference D-2):

Pressure	Design Limit DNBR
400 to 700 psia	1.59
700 to 1000 psia	1.20

TABLE D-3 Oconee SCD Statepoints

Statepoint Number	Power[1] (% RTP)	RCS Flow[2] % DF	Pressure (psia)	Core Inlet Temperature (°F)	Axial Peak (F_z @ Z)	Radial Peak $F\Delta H$
1						
2						
3						
4						
5						
6						
7						
8						
9						
10						
11						
12						
13						
14						
15						
16						
17						
18						
19						
20						
21						
22						
23						
24						
25						
26						
27						
TR22						

1) 100% RTP = 2568 Megawatts Thermal
2) 100% design flow is equal to 352,000gpm.

TABLE D-4 Oconee Statistically Treated Uncertainties

Parameter	Type	Type of Distribution	Uncertainty	Standard Deviation
Reactor System				
Core Power*	Measurement	Normal	+/-2.0%FP	+/-1.0%FP
Core Flow	Measurement	Normal	4 Pump: +/-2.0%	+/-1.0%
			3 Pump: +/-3.2%	+/-1.6%
			2 Pump: +/-4.2% design	+/-2.1% design
Pressure	Measurement	Normal	+/-30.0 psi	+/-15.0 ps
Temperature	Measurement	Normal	+/-2.0°F	+/-1.0°F
Nuclear				
FΔH	Calculation	Normal	---	+/-2.84%
Fz	Calculation	Normal	---	+/-2.91%
Z	Calculation	Uniform	+/-6 inches	---
Fq	Calculation	Normal	[	]
Hot Channel Flow Area	Measurement	Uniform	[ ]	---
DNBR	Correlation	Normal	---	9.268%
DNBR	Code	Normal	[	]

* Percentage of 100% RTP (69.75 MWth wherever applied).

TABLE D-4 Continued   Oconee Statistically Treated Uncertainties

Parameter	Justification
System Pressure	This uncertainty accounts for random uncertainties in various instrumentation components. Since the random uncertainties are normally distributed, the square root of the sum of the squares (SRSS) that results in the pressure uncertainty is also normally distributed.
Inlet Temperature	Same approach as Pressure uncertainty.
Core Power	The core power uncertainty was calculated by statistically combining various random uncertainties associated with the measurement of core power. Since the random uncertainties are normally distributed, the SRSS that results in the core power uncertainty is also normally distributed.
Core Flow	Same approach as Core Power uncertainty.
Radial Power, $F_{\Delta H}$	This uncertainty accounts for the error associated in the physics code's calculation of radial assembly power and the measurement of the assembly power. This uncertainty distribution is normal.
Axial Peak Power, F_Z	This uncertainty accounts for the axial peak prediction uncertainty of the physics codes. The uncertainty is normally distributed.
Axial Peak Location, Z	This uncertainty accounts for the possible error in interpolating on axial peak location in the maneuvering analysis. The uncertainty is one half of the physics code's axial node. The uncertainty distribution is conservatively applied as uniform.

TABLE D-4 Continued   Oconee Statistically Treated Uncertainties

Parameter	Justification
Rod Power HCF, Fq	This uncertainty accounts for the increase in rod power due to manufacturing tolerances. The uncertainty in calculating the peak pin from assembly radial peak is also statistically combined with the manufacturing tolerance uncertainty to arrive at the correct value. The uncertainty is normally distributed and conservatively applied as one-sided in the analysis to assure the MDNBR channel location is consistent for all cases.
Hot Channel Flow Area	This uncertainty accounts for manufacturing variations in the instrument guide tube subchannel flow area. This uncertainty is uniformly distributed and is conservatively applied as one-sided in the analysis to ensure the MDNBR channel location is consistent for all cases.
DNBR - Correlation	This uncertainty accounts for the CHF correlation's ability to predict DNB. The uncertainty distribution is applied as normally distributed.
Code/Model	This uncertainty accounts for the thermal-hydraulic code uncertainties and offsetting conservatisms. This uncertainty also accounts for the small DNB prediction differences between the various model sizes. The uncertainty distribution is normally distributed.

TABLE D-5 Oconee Statepoint Statistical Results

BWU-Z Critical Heat Flux Correlation With Performance Factor

500 Case Runs

Statepoint #	Mean	σ	Coefficient of Variation	Statistical DNBR
1				
2				
3				
4				
5				
6				
7				
8				
9				
10				
11				
12				
13				
14				
15				
16				
17				
18				
19				
20				
21				
22				
23				
24				
25				
26				
27				
TR22				

TABLE D-5 Continued   Oconee Statepoint Statistical Results

BWU-Z Critical Heat Flux Correlation With Performance Factor

5000 Case Runs

Statepoint #	Mean	σ	Coefficient of Variation	Statistical DNBR
1-T				
3-T				
17-T				
21-T				
22-T				
24-T				
TR22-T				

TABLE D-6 Oconee Key Parameter Ranges

Parameter	Maximum	Minimum
Core Power (%RTP)		
Pressure (psia)		
T inlet (deg F)		
RCS Flow (% Design)		
FΔH, Fz, Z		

All values listed in this table are based on the currently analyzed Statepoints. Ranges are subject to change based on future statepoint conditions.

REFERENCES

D-1. DPC-NE-2003P-A, Oconee Nuclear Station Core Thermal-Hydraulic Methodology Using VIPRE-01, October 1989.

D-2. The BWU Critical Heat Flux Correlations, Addendum 1 to BAW-10199P-A, Framatome Cogema Fuels, Lynchburg, Virginia, April 6, 2000.

D-3. VIPRE-01: A Thermal-Hydraulic Code For Reactor Cores, EPRI NP-2511-CCM-A, Vol. 1-4, Battelle Pacific Northwest Laboratories, August 1989.

D-4 DPC-NE-2005P-A, Rev. 1, Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology, Appendix C, November 1996.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 1, 1998

RECEIVED
JUL 7 1998
DUKE POWER CO.
NUCLEAR ENGINEERING

Mr. William R. McCollum
Vice President, Oconee Site
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SUBJECT: REQUEST FOR ADDITIONAL INFORMATION - APPENDIX D, "OCONEE PLANT SPECIFIC DATA, MARK-B11 FUEL, APPLICATION OF BWU-Z CHF CORRELATION TO MARK-B11 MIXING VANE SPACER GRID FUEL DESIGN," TO DPC-NE-2005P, "DUKE POWER COMPANY THERMAL-HYDRAULIC STATISTICAL CORE DESIGN METHODOLOGY" (TAC NOS. 98660, 98661, AND 98662)

Dear Mr. McCollum:

By letter dated April 22, 1997, Duke Energy Corporation transmitted the subject topical report, Appendix D to DPC-NE-2005P, for staff review. In order to complete its review, the NRC staff has determined that additional information is needed. The staff's request for additional information is enclosed.

Sincerely,

David E. LaBarge, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-269, 50-270, and 50-287

Enclosure: Request for Additional Information

cc w/encl: See next page

Oconee Nuclear Station

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Request for Additional Information
Review of Duke Topical Report DPC-NE-2005P,
Appendix D, "Oconee Plant Specific Data, Mark-B11 Fuel,
and Application of BWU-Z CHF Correlation to Mark-B11
Mixing Vane Spacer Grid Fuel Design"
to DPC-NE-2005P, "Duke Power Company
Thermal-hydraulic Statistical Core Design Methodology"

1. The safety evaluation report (SER) for DPC-NE-2005P-A requires that in all applications of the statistical core design methodology, the uncertainties and distributions used in the analysis will be justified on a plant-specific basis. This has not been done in Appendix D, which presents plant-specific data for Oconee with Mark-B11 fuel. Comparing Table D-4 of Appendix D with Table A-2 of Appendix A (which contains the approved uncertainties and distributions for the Oconee units with Mark-B10 fuel), there are four major changes, none of which are explained adequately. Specifically:

   (a) The core flow uncertainty has been increased from ±2.0 percent design (with standard deviation ±1.0 percent design) to ±4.2 percent design (with standard deviation ±2.1 percent design). The report says simply that the value was increased "to ensure that it is bounding." Bounding in what way? How was this determined? It appears to be an arbitrary adjustment to what should be a real indicator of the uncertainty in the measured core flow. What is the justification for this change?

   (b) Table A-2 includes the parameter Fq" (local heat flux hot channel flow (HCF)), which is an uncertainty to account for the decrease in departure from nuclear boiling ratio (DNBR) at the point of minimum DNBR due to engineering tolerances. It also accounts for flux depression at a spacer grid, and has a value of +2.08 percent (with standard deviation +1.26 percent). This parameter has been omitted from Table D-4. What is the justification for this change?

   (c) Table A-2 includes the parameter Fq (rod power HCF), which is an uncertainty to account for rod power increases due to manufacturing tolerances. This parameter also includes the uncertainty in calculating the pin peak from the assembly radial peak, and has a value of +2.27 percent (with standard deviation +1.38 percent) for Mark-B10 fuel. In Table D-4, this parameter has the value +2.40 percent (with standard deviation +1.46 percent) for Mark-B11 fuel. How was this uncertainty determined, and why is it larger for Mark-B11 fuel than for Mark-B10 fuel?

   (d) The HCF area uncertainty is reported as -3.00 percent in Table D-4, unchanged from the value in Table A-2 for Mark-B10 fuel, even though there are significant differences in the assembly geometry of Mark-B11 fuel. In addition, the value reported for the parameter Fq indicates that there are significant differences in the manufacturing tolerances for the fuel rods in Mark-B11 fuel, which would seem to imply that there should also be significant differences in the flow channel geometry variations. What is the justification for using the value of -3.00 percent for this parameter?

Enclosure

2. The description of how transition cores will be treated is unclear. Please provide additional information, addressing the following points:.

   (a) What is a "transition core penalty," and how is it determined?

   (b) The local pressure drop differences between the Mark-B10 and Mark-B11 fuel assemblies mean that the local assembly flow distributions may be very different in a mixed core, due to differences in inter-assembly crossflow patterns. The departure from nucleate boiling (DNB) behavior of a mixed core may, therefore, be significantly different from that of a full core of Mark-B11 fuel only. Justify the assumption that the BWU-Z CHF correlation can be applied to Mark-B11 fuel in a core containing both Mark-B10 and Mark-B11 fuel.

   (c) Two options are described (see p. D-7) that will be used to "conservatively compensate" for the transition core penalty. The report states that they will be applied "as necessary" to determine the DNB effect of a transition core. What are the criteria for selecting one or the other of the two options? How will it be determined that the selected option is "conservative" for a given transition core?

   (d) One option of the two described on p. D-7 is to explicitly apply a penalty to the Mark-B11 fuel generic peaking limit based on a full Mark-B11 core. What is this penalty? How is it determined? How will it be determined that the penalty adequately accounts for the effects of a mixed core on DNB behavior?

3. Table D-2 (p. D-14) claims a pressure range of 400 to 2465 psia for the BWU-Z correlation with the Mark-B11V multiplier. The database supporting this form of the correlation includes tests only over the pressure range 695 to 2425 psia. In addition, there is a distinct nonconservative bias evident in the correlation's predictions with decreasing pressure (see Figure D-3, p. D-11). The BWU-Z correlation for Mark-BW17 fuel (as documented in BAW-10199-A) has a demonstrated bias with decreasing pressure, and the SER for this correlation specifies a separate design limit DNBR of 1.59 for pressures below 700 psia. If the BWU-Z correlation with the Mark-B11V multiplier is to be applied to conditions where the pressure is below 700 psia, what value will be used for the design limit DNBR and how will it be determined?

4. The SER for DPC-NE-2005P-A requires that the selected state points for an application of the SCD methodology shall be justified to be appropriate, on a plant-specific basis. Documentation of this justification in Appendix D consists only of the statement on p. D-1, "...state point ranges were selected to bound the unit and cycle-specific values of the Oconee Station." However, the document also notes that the values of the key parameter ranges used to define the state points (in Table D-6, p. D-21) are "based on the currently analyzed state points," and further notes that "ranges are subject to change based on future state point conditions." The procedure and justification for selecting state points is unclear, and additional information is needed. Specifically, please provide a more detailed description of how the state points are selected for the Oconee plant-specific data, with particular attention to how bounding values are to be determined for the B11 and mixed B10/B11 cores.

5. The calculations with the VIPRE-01 code using the BWU-Z correlation form for B11 fuel show essentially the same results as the those obtained with LYNX over the correlation's database (as documented in Addendum 1 of BAW-10199). However, the BWU-Z correlation as modified for analysis of B11 fuel has not yet been approved by the staff, and the topical report describing this correlation, Addendum 1 of BAW-10199, is still being reviewed. This means that the design limit DNBR for the parameter ranges stated in Table D-2 may not be the final approved value or range of applicability. Specifically, the database for the form of the correlation spans a pressure range of 700 to 2400 psia, not the 400 to 2400 psia range stated in Table D-2. Also, the plot in Figure D-3 (see p. D-11) shows a distinct nonconservative bias with decreasing pressure (which is identical to the trend shown for the correlation in the Addendum 1 submittal). There is also a nonconservative bias with increasing power, clearly shown by the plot of measured versus predicted Critical Heat Flux (CHF) in Figure D-1. What would be the effect on the thermal hydraulic statistical core design analysis for Oconee if the DNBR design limit of the CHF correlation for B11 fuel were to be increased, or if the range of applicability of the correlation were to be limited to pressures of 700 to 2400 psia?

Duke Energy

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September 21, 1998

U. S. Nuclear Regulatory Commission
Washington, D. C. 20555-0001
Attention: Document Control Desk

Subject: Duke Energy Corporation

Oconee Nuclear Station, Units 1, 2 and 3
Docket Numbers 50-269, 50-270, and 50-287

Response to NRC Request for Additional Information on Appendix D to Topical Report DPC-NE-2005-P, "Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology"

This submittal contains information that Duke Energy Corporation considers PROPRIETARY and is being made pursuant to 10CFR 2.790.

By letter dated July 1, 1998 the NRC requested additional information on Appendix D to Topical Report DPC-NE-2005P, "Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology." This topical report had been previously submitted for NRC review by Duke letter dated April 22, 1997.

The questions contained in the July 1 NRC letter, and the corresponding Duke answers, are provided in the attachment to this letter. Additionally, Table D-1, which is also included in the attachment, has been revised to correct a typographical error.

Some of the information contained in the attachment is considered proprietary. In accordance with 10CFR 2.790, Duke Energy Corporation requests that this information be withheld from public disclosure. An affidavit which attests to the proprietary nature of the affected information is included with this letter. A non-proprietary version of the affected material is also included.

Please address any comments or questions regarding this matter to J. S. Warren at (704) 382-4986.

Very truly yours,

M. S. Tuckman

M. S. Tuckman

Attachments

xc:

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ELL

AFFIDAVIT OF M. S. TUCKMAN

1. I am Executive Vice President of Duke Energy Corporation; and as such have the responsibility for reviewing information sought to be withheld from public disclosure in connection with nuclear power plant licensing; and am authorized on the part of said Corporation (Duke) to apply for this withholding.

2. I am making this affidavit in conformance with the provisions of 10CFR 2.790 of the regulations of the Nuclear Regulatory Commission (NRC) and in conjunction with Duke's application for withholding, which accompanies this affidavit.

3. I have knowledge of the criteria used by Duke in designating information as proprietary or confidential.

4. Pursuant to the provisions of paragraph (b)(4) of 10CFR 2.790, the following is furnished for consideration by the NRC in determining whether the information sought to be withheld from public disclosure should be withheld.

   (i) The information sought to be withheld from public disclosure is owned by Duke and has been held in confidence by Duke and its consultants.

   (ii) The information is of a type that would customarily be held in confidence by Duke. The information consists of analysis methodology details, analysis results, supporting data, and aspects of development programs relative to a method of analysis that provides a competitive advantage to Duke.

M. S. Tuckman

M. S. Tuckman

(Continued)

(iii) The information was transmitted to the NRC in confidence and under the provisions of 10CFR 2.790, it is to be received in confidence by the NRC.

(iv) The information sought to be protected is not available in public to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is marked in the attachment to Duke Energy Corporation letter dated September 21, 1998; SUBJECT: Response to NRC Request for Additional Information on Topical Report DPC-NE-2005P, "Duke Power Company Thermal-Hydraulic Statistical Core Design Methology." This information enables Duke to:

(a) Respond to Generic Letter 83-11, "Licensee Qualification for Performing Safety Analyses in Support of Licensing Actions."

(b) Support Facility Operating Licenses/Technical Specifications amendment requests for Babcock & Wilcox PWRs.

(c) Perform safety evaluations per 10CFR50.59.

(vi) The proprietary information sought to be withheld from public disclosure has substantial commercial value to Duke.

(a) It allows Duke to reduce vendor and consultant expenses associated with supporting the operation and licensing of nuclear power plants.

M. S. Tuckman

M. S. Tuckman

(Continued)

(b) Duke intends to sell the information to nuclear utilities, vendors, and consultants for the purpose of supporting the operation and licensing of nuclear power plants.

(c) The subject information could only be duplicated by competitors at similar expense to that incurred by Duke.

5. Public disclosure of this information is likely to cause harm to Duke because it would allow competitors in the nuclear industry to benefit from the results of a significant development program without requiring commensurate expense or allowing Duke to recoup a portion of its expenditures or benefit from the sale of the information.

M. S. Tuckman, being duly sworn, states that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth within are true and correct to the best of his knowledge.

M. S. Tuckman

M. S. Tuckman, Executive Vice President

Subscribed and sworn to before me this 22nd day of

September, 1998

Linda Case Smith

~~Mary P. Nelms~~, Notary Public
Linda Case Smith

My Commission Expires: ~~January 22, 2001~~ May 6, 2000

SEAL

NRC Questions On Mark-B11 SCD Submittal

Questions shown in italics, answers immediately follow.

1. *The safety evaluation report (SER) for DPC-NE-2005P-A requires that in all applications of the statistical core design methodology, the uncertainties and distributions used in the analysis will be justified on a plant-specific basis. This has not been done in Appendix D, which presents plant-specific data for Oconee with Mark-B11 fuel. Comparing Table D-4 of Appendix D with Table A-2 of Appendix A (which contains the approved uncertainties and distributions for the Oconee units with Mark-B10 fuel), there are four major changes, none of which are explained adequately. Specifically:*

   a) *The core flow uncertainty has been increased from +/- 2.0 percent design (with standard deviation of +/-1.0 percent design) to +/-4.2 percent design (with standard deviation +/-2.1 percent design). The report says simply that the value was increased "to ensure that it is bounding." Bounding in what way? How was this determined? It appears to be an arbitrary adjustment to what should be a real indicator of the uncertainty in the measured flow. What is the justification for this change?*

The current Chapter 15 analyses for Oconee were performed by FCF. Duke Power has recently reanalyzed the Chapter 15 transients and submitted a topical report that is currrently being reviewed (DPC-NE-3005). As part of this effort, Duke recalculated the flow uncertainties for combinations of 4, 3, and 2 operating reactor coolant pumps. Table D-4 has been revised to show the flow uncertainties used in the BWU-Z SCD analyses (the original table only listed the maximum flow uncertainty for 2 pump operation). The statepoints listed in Table D-3 were propagated using the appropriate flow uncertainty. For example, statepoint 22 is the limiting statepoint for the 2 pump coastdown transient. Thus, this statepoint was propagated using a flow uncertainty of 4.2 % (std. deviation of 2.1 %).

Statepoints using the flow uncertainties for 3 and 2 operating reactor coolant pumps have also been propagated using the BWC correlation. The statistical DNB limit for all cases using the higher flow uncertainties was less than the SCD limit given in Appendix A. Thus, no NRC submittal was required based on the criteria given in Table 7 of DPC-NE-2005.

b) Table A-2 includes the parameter Fq" (local heat flux hot channel flow (HCF)), which is an uncertainty to account for the decrease in departure from nuclear boiling ratio (DNBR) at the point of minimum DNBR due to engineering tolerances. It also accounts for flux depression at a spacer grid, and has a value of [ ] (with standard deviation [ ]). This parameter has been omitted in Table D-4. What is the justification for this change?

The local heat flux hot channel factor accounts for the effects on DNBR of local variations in pellet enrichment and weight on local (hot spot) power, and flux depressions at spacer grid. Small local heat flux spikes have been shown to have no effect on the critical heat flux (CHF) per the Oconee 1 Cycle 14 Reload Report, DPC-RD-2018.

DPC-RD-2018 was submitted as supplementary information in support of Technical Specification changes for Oconee Unit 1 (Amendment 191, TAC 80378), Unit 2 (Amendment 191, TAC 80379), and Unit 3 (Amendment 188, TAC 80380). NRC approved the Technical Specification change. Duke considered the NRC's implementation of the recommended Technical Specification changes to be implicit approval of DPC-RD-2018.

Removal of Fq" from DNBR analyses was justified in DPC-RD-2018 based on WCAP-8202 and CENPD-207. The WCAP-8202 evaluation concluded that the data and analysis clearly indicate no effect on the minimum DNBR due to large local heat flux spikes. The spikes tested were in the region of MDNBR and were 20% greater than the heat flux in the immediate vicinity. The conclusion of these reports is that the local heat flux spikes associated with fuel densification have no affect on DNBR. This effect is generic to PWR fuel types and was confirmed to be applicable by the fuel vendor. Additionally, the magnitude of Fq" calculated by the vendor for Mark-B11 fuel is much smaller, [ ]. Based on this information and the approved reload report submittal, the Fq" factor was omitted from the Mark-B11 analyses and, therefore, Table D-4.

c) Table A-2 includes the parameter F_q (rod power HCF), which is an uncertainty to account for rod power increases due to manufacturing tolerances. This parameter also includes the uncertainty in calculating the pin peak from the assembly radial peak, and has a value of [ ] (with standard deviation [ ]) for Mark-B10 fuel. In Table D-4, this parameter has the value of [ ] (with standard deviation of [ ]) for Mark-B11 fuel. How was this uncertainty determined, and why is it larger for Mark-B11 fuel than for Mark-B10 fuel?

A rod power hot channel factor of [ ] was specified in the original issue of DPC-NE-2005 for Mark-B10 fuel. This value has increased to [ ] for the Mark-B10 fuel (beginning with Oconee 1, 2, and 3 Batch 17) and [ ] for the Mark-B11 fuel to account for dry blending of UO_2 powder to achieve the desired enrichment. The Statistical Design Limit (SDL) given in DPC-NE-2005 was shown to still be valid for Mark-B10 fuel using the increased rod power hot channel factor evaluated as per the process described in Table 7 of DPC-NE-2005P-A.

The value for F_q used in the SCD analysis is calculated as follows based on the rod power hot channel factor of [ ] for Mark-B11 provided by the fuel manufacturer and the radial peak uncertainty of [ ] per DPC-NE-1004P-A.

$$F_q\% = [\quad]$$

The standard deviation is calculated as follows:

$$\sigma(F_q)\% = \frac{[\quad]}{1.645}$$

$$= [\quad]$$

The 1.645 is the one-sided 95/95 statistical K factor for an infinite number of points.

The rod power hot channel factor, Fq, is provided as part of the fuel fabrication process and is formally transmitted in batch specific design and fabrication data supplied by the fuel manufacturer for each reload batch. If the rod power hot channel factor were greater than [ ], then the fuel manufacturer would notify Duke Power and the impact on the SDL will be evaluated.

d) The HCF area uncertainty is reported as [ ] in Table D-4, unchanged from the value in Table A-2, for Mark-B10 fuel, even though there are significant differences in the assembly geometry of Mark-B11 fuel. In addition, the value reported for the parameter Fq indicates that there are significant differences in the manufacturing tolerances for the fuel rods in Mark-B11 fuel, which would seem to imply that there should also be significant differences in the flow channel geometry variations. What is the justification for using the value of [ ] for this parameter?

The value listed in Table D-4 was provided by the fuel manufacturer. The fuel fabricator will verify through inspection of the final fuel assemblies and components that the uncertainty on flow area assumed in the analysis is valid. The same inspection techniques employed in earlier designs will be used for the Mark-B11 fuel. Comparison to acceptance criteria for the Mark-B11 fuel will ensure compliance with the [ ] flow area uncertainty. Water channel data taken on the Mark-B11 lead test assemblies were evaluated and found to be acceptable. If the flow area uncertainty is greater than [ ], then the fuel manufacturer will notify Duke Power and the impact on the SDL will be evaluated as per the process described in Table 7 of DPC-NE-2005P-A.

2. *The description of how transition cores will be treated is unclear. Please provide additional information, addressing the following points:*

   a) What is a "transition core penalty," and how is it determined?

A generic transition core penalty is determined by comparing the DNBR results from a full core of Mark-B11 fuel with the DNBR results from a conservative Mark-B11/Mark-B10 transition core. The 9 channel transition core model licensed in DPC-NE-2003 and described in Appendix D of DPC-NE-2005 is used in this analysis. The MDNBR (or allowable radial peaking) is calculated with both models for a range of fluid conditions and axial peaking combinations. The largest penalty calculated from this matrix of conditions is used as the transition core penalty.

The process for determining a generic transition core penalty is as follows:

1. Develop an 8 channel Mark-B11 full core model and a 9 channel Mark-B11/Mark-B10 transition core model (per DPC-NE-2003P-A and described in Appendix D of DPC-NE-2005).

2. Evaluate each model for a range of fluid conditions (shown below) that is representative of fluid conditions for which the Maximum Allowable Peaking limits are developed using VIPRE-01. These fluid conditions are evaluated at the axial peaking conditions shown below.

Parameter	**Maximum**	**Minimum**
Core Power (% RTP)	110	80
RCS Flow (% Design Flow)	107.5	80.3
T inlet (deg F)	572.8	529.2
Pressure (psia)	2242	1830
Fz (normalized axial peak)	1.1, 1.4, 1.7, 2.1	
z (location of axial peak)	0.2, 0.4, 0.6, 0.8	

The radial peaking results from VIPRE-01, for both the Mark-B11 full and transition core models at each fluid condition, were compared to determine the limiting fluid condition. Then a complete set of MAP curves were developed for both the full core Mark-B11 and the transition core models. These peaking results were compared, and a maximum transition core peaking penalty was determined. In addition, the axial dependency of the transition core penalty was determined. The response to question 2c) specifies the options for the application of the transition core penalty.

b) *The local pressure drop differences between the Mark-B10 and Mark-B11 fuel assemblies mean that the local assembly flow distributions may be very different in a mixed core, due to differences in inter-assembly crossflow patterns. The departure from nucleate boiling (DNB) behavior of a mixed core may, therefore, be significantly different from that of a full core of Mark-B11 fuel only. Justify the assumption that the BWU-Z CHF correlation can be applied to Mark-B11 fuel in a core containing both Mark-B10 and Mark-B11 fuel*

In mixed cores, the possibility of large axial velocity upsets at or around dissimilar grids exists. These upsets imply different local thermal-hydraulic conditions in surrounding subchannels. It has been questioned as to whether traditional steady state CHF correlations are applicable in this instance.

The FCF CHF correlation form (BWU) is composed of three parts: a uniform part dependent solely on the local thermal-hydraulic conditions of pressure, mass velocity and thermodynamic quality at the axial location of CHF, a non-uniform F factor modification dependent on the shape of the axial heat flux input, and a multiplicative geometric factor dependent on the overall fuel assembly grid spacing and heated length. It is with the uniform, local conditions part that the mixed core conditions question surfaces.

CHF correlations are developed from data from full length electrically heated bundles in 5-by-5 rod arrays. For each data point, the inlet conditions of coolant mass velocity, pressure and temperature are known, as is the power (heat flux) required to produce a DNB event. The local thermal-hydraulic conditions at the axial location of CHF must then be calculated with a computer code.

The proof of applicability of a CHF correlation, then, is how well it can predict the critical heat flux that was measured in the DNB event using the calculated local conditions. Thus, the applicability of a CHF correlation is dependent not only on its form and data base, but on the accuracy with which the local conditions can be calculated in any given situation. Because of the size of the test section (a 5-by-5 rod array) and the use of a series of single spacer grids (axially), normal CHF tests do not exhibit large hydraulic axial differences. FCF, however, has performed one test with widely varying subchannel axial resistances producing the large velocity upsets representative of mixed core conditions. This test was a 5-by-5 test of the Mark B zircaloy grid modeled as the corner intersection of four fuel assemblies. LDV testing of the intersection grid showed velocity depressions as large as 50% between the intersection subchannel and the surrounding unit cell subchannels. This test was conducted at the Babcock & Wilcox Alliance Research Center and is documented in BAW-10143P-A (BWC correlation of Critical Heat Flux, April, 1985). In the topical, the measured to predicted (M/P) CHF results were compared for two traditional test bundles and the intersection bundle. The guide tube bundle (B15) had an average M/P of 0.971, the unit cell bundle (B16) 0.985 and the intersection bundle (B17) 0.976. The difference in M/P results is statistically insignificant. This qualified the BWC correlation for use with the Mark B fuel assembly design.

The local conditions necessary for the BWC correlation were calculated with a thermal-hydraulic computer code. The local conditions for the normal unit and guide tube bundles had very little axial upset, while the intersection bundle (which produces conditions representative of a mixed core) had severe upsets resulting from the two to one velocity upsets. The fact that the BWC correlation performed consistently on conditions representative of both homogeneous and mixed cores confirms that the FCF local conditions CHF correlations are valid for both homogeneous and mixed core applications as long as the local conditions can be accurately predicted by the subchannel thermal-hydraulic computer code.

In this particular application, the velocity upsets calculated in the Mark-B11 transition core analysis are on the order of 10%. These calculations assume the limiting geometry (a single Mark-B11 assembly surrounded by Mark-B10 fuel). Since the test data included local depressions as large as 50%, the FCF test results bound by a significant amount the transition core configuration.

c) Two options are described (see p. D-7) that will be used to "conservatively compensate" for the transition core penalty. The report states that they will be applied "as necessary" to determine the DNB effect of a transition core. What are the criteria for selecting one or the other of the two options? How will it be determined that the selected option is "conservative" for a given transition core?

The three methods for penalizing a transition core are to

1) Penalize the DNBR limit used in the analyses directly or
2) Penalize the Maximum Allowable Peaking Total (MATP) limits determined for a transition core
3) Use a combination of the two above.

The penalty applied using either method 1 or 2 is based on the most limiting transition core statepoint determined as described in the response to Question 2(a) above. This ensures either option is conservative for the transition core.

Option 3) listed above is the current one selected to provide a bounding, conservative transition core penalty while maximizing core design flexibility. As described in Question 2(a), the transition core penalty was evaluated with a subset of axial peak locations (F_Z, Z) over a wide range of fluid conditions. Then, the fluid condition with the largest penalty was evaluated with a complete set of axial peaks (F_Z from 1.1 to 2.1, Z from 0.01 to 1.0). This is the same set of axial peak locations used to generate the MATP limits and resulting curves described in DPC-NE-2003.

In this analysis, the transition core penalty shows axial shape dependence. Due to this relationship, it is reasonable to include part of the penalty directly in the applicable MATP limits. Based on the analysis described above, the transition core penalty is applied as follows:

1. A 0.5% peaking penalty (1.5% DNB penalty) is applied to the retained DNB margin available between the SDL and the DDL for Mark-B11 transition cores. This directly applies the penalty to all DNB calculations.

2. A 1% radial peaking penalty is applied to selected axial peak locations. These are generally the large axial peaks (F_Z of ≥ 1.4) in the top half of the core (at $Z \geq 0.6$). Again, these axial peak locations were determined by comparison of a complete set of MATP curves. This penalty will be applied to all Mark-B11 MATP's limits at the axial peaking locations necessary.

Duke will retain the option of applying any of the three methods described as a conservative transition core penalty such that cycle design impact is minimized.

d) One option of the two described on p. D-7 is to explicitly apply a penalty to the Mark-B11 fuel generic peaking limit based on a full Mark-B11 core. What is the penalty? How is it determined? How will it be determined that the penalty adequately accounts for the effects of a mixed core on DNB behavior?

The transition core penalty is determined as described in the response to Question 2(a) above. This adequately accounts for the mixed core effect as explained in the response to Question 2(b). As stated in the response to Question 2(c), the transition core penalty can be applied to the maximum allowable peaking (MAP) limits calculated for a full Mark-B11 core. This reduces the allowable peaking in the transition core to account for the hydraulic and geometry effects. This also ensures that the MDNBR in all transition core analyses is greater than the licensed SDL.

As with previously licensed transition core methods, the transition core geometry for a reload cycle can be specifically modeled using the 64 channel model described in DPC-NE-2003. This larger model allows analyses of the actual cycle loading pattern to determine the impact of a mixed core on the maximum allowable peaking limits for the transition cycle.

3. Table D-2 (p. D-14) claims a pressure range of 400 to 2465 psia for the BWU-Z correlation with the Mark-B11V multiplier. The database supporting this form of the correlation includes tests only over the pressure range 695 to 2425 psia. In addition, there is a distinct nonconservative bias evident in the correlation's predictions with decreasing pressure (See Figure D-3, p. D-11). The BWU-Z correlation for Mark-BW17 fuel (as documented in BAW-10199-A) has demonstrated bias with decreasing pressure, and the SER for this correlation specifies a separate design limit DNBR of 1.59 for pressures below 700 psia. If the BWU-Z correlation with the Mark-B11V multiplier is to be applied to conditions where the pressure is below 700 psia, what value will be used for the design limit DNBR and how will it be determined?

See Question 5 for response.

4. *The SER for DPC-NE-2005P-A requires that the selected state points for an application of the SCD methodology shall be justified to be appropriate, on a plant-specific basis. Documentation of this justification in Appendix D consists only of the statement on p. D-1 "...state point ranges were selected to bound the unit and cycle-specific values of the Oconee Station." However, the document also notes that the values of key parameter ranges used to define the state points (Table D-6, p. D-21) are "based on the currently analyzed state points," and further notes that "ranges are subject to change based on future state point conditions." The procedure and justification for selecting state points is unclear, and additional information is needed. Specifically, please provide a more detailed description of how the state points are selected for the Oconee plant-specific data, with particular attention to how bounding values are to be determined for B11 and mixed B10/B11 cores.*

The power/flow/pressure/temperature ranges for the SCD analyses are determined by the steady state and transient analyses for which DNBR is calculated. The Safety Analysis group provides the statepoint conditions to be evaluated in the SCD analysis. These statepoints represent expected ranges of operation in Chapter 15 transients. The statepoints shown in Table D-6 currently bound the range of conditions for Oconee where the SCD methodology is used to calculate DNBR. As necessary, additional statepoints from Safety Analysis are evaluated using the approved methodology in DPC-NE-2005 to verify that the Statistical DNB Limit determined is still bounding for the new set of conditions.

5. *The calculation with the VIPRE-01 code using BWU-Z correlation form for B11 fuel show essentially the same results as those obtained with LYNX over the correlation's database (as documented in Addendum 1 of BAW-10199). However, the BWU-Z correlation as modified for analysis of B11 fuel has not yet been approved by the staff, and the topical report describing this correlation, Addendum 1 of BAW-10199, is still being reviewed. This means that the design limit DNBR for the parameter ranges stated in Table D-2 may not be the final approved value or range of applicability. Specifically, the database for the form of the correlation spans a pressure range of 700 to 2400 psia, not 400 to 2465 psia range stated in Table D-2. Also, the plot in Figure D-3 (see p. D-11) shows a distinct nonconservative bias with decreasing pressure (which is identical to the trend shown for the correlation in the Addendum 1 submittal). There is also a nonconservative bias with the increasing power, clearly shown by plot of measured versus predicted Critical Heat Flux (CHF) in Figure D-1. What would be the effect on the thermal-hydraulic statistical core design analysis for Oconee if the DNBR design limit of the CHF correlation for B11 fuel were to be increased, or if the range of applicability of the correlation were to be limited to pressures of 700 to 2400 psia?*

The pressure range reported in Table D-2 is consistent with the conclusion made in Addendum 1 of BAW-10199. The Addendum 1 conclusion states that the correlation parameter range for the BWU-Z correlation with the Mark-B11V multiplier is the same as the BWU-Z correlation. The data base for the Mark-B11 fuel included a pressure range of 595 psia to 2425 psia as stated in Table E-7 of Addendum 1 to BAW-10199. Also, Figure D-3 shows a slight conservatism with decreasing pressure. Likewise, Figure D-1 shows a slight conservatism with increasing power.

The pressure range for the statepoints evaluated in Appendix D is 1600 psia to 2242 psia. The pressure/temperature conditions for these statepoints were selected to bound the range of fluid conditions at Oconee which will use the statistical DNBR methodology. Other DNB calculations are performed via the non-statistical DNB method. Non-statistical DNB calculations will use the applicable design limit DNBR (from the approved BWU-Z correlation, see table below). The correlation design limit DNBR (1.19) applies only at or above a nominal pressure of 1000 psia (Reference D-5 of Appendix D). In the lower pressure region (below a nominal pressure of 1000psia) the design limit DNBR in the following table will be used (Reference D-5):

Pressure	Design Limit DNBR
400 to 700 psia	1.59
700 to 1000 psia	1.20

Attached is Table D-2 which has been updated to clarify the pressure dependency of the design limit DNBR. Also, references D-2 and D-5 have been updated to reflect the current revision of the approved topicals.

If a statepoint with pressure less than 1600 psia were identified, it would be propagated using the applicable CHF correlation standard deviation. A statepoint with pressure less than 1000 psia is not expected for Oconee SCD analyses. If a statepoint with a pressure less than 1000 psia were analyzed, the applicable design limit DNBR will be used and the impact of the higher correlation standard deviation on the statistical design limit would be directly calculated. This verifies the statistical design limit for the statepoint is bounded. If the SDL for the new statepoint is greater than the licensing limit, the higher SDL will be used when analyzing the lower pressure conditions. This is in accordance with the methodology as described in Table 7 of DPC-NE-2005.

Any changes to the CHF correlation or restrictions in its application resulting from the NRC review process will be communicated to Duke Power by the fuel vendor. If the Mark-B11 CHF correlation range of applicability is changed, the SCD analysis would be revised as needed to reflect the modification. The correlation will not be used for DNB calculations outside the parameter range stated in the approved correlation topical. If the correlation standard deviation increases above the value used in the analyses, the limiting statepoint will be re-propagated to verify the SDL given in Appendix D.

TABLE D-1 MARK-B11 FUEL ASSEMBLY DATA

(TYPICAL)

GENERAL FUEL SPECIFICATIONS

Fuel rod diameter, in. (Nom.)	0.416
Thimble tube diameter, in. (Nom.)	0.530
Instrument guide tube diameter, in. (Nom.)	0.554(1)/0.567(2)
Fuel rod pitch, in (Nom.)	0.568
Fuel assembly pitch, in. (Nom.)	8.587
Fuel rod length, in. (Nom.)	154.16

(1) Above lowest mixing vane grid (MV) and between MV grids.

(2) Below the first mixing vane grid and above the top of the last mixing vane.

GENERAL FUEL CHARACTERISTICS

Grids:	Material	Quantity	Location	Type
	Inconel	2	Upper and Lower	Non-Mixing Vane
	Zircaloy	6	Intermediate	1 Non-Mixing Vane, 5 Mixing Vane

Fuel Rods:	Material	Quantity
	Zircaloy-4	208

Fuel Cycle Design Assembly Features

Fuel Assy.	Mark
Designation:	B11
Features: vane	Smaller clad outside diameter and mixing grids.

TABLE D-2 VIPRE-01 BWU-Z Correlation with Mark-B11V Multiplier Verification

CHF Test Database Analysis Results

VIPRE-01/LYNXT Statistical Results

	VIPRE-01	LYNXT
n, # Of data	216	216
N, degrees of freedom (n-1)	215	215
M/P, Average measured to predicted CHF	1.0084	1.0040
σ (M/P/N)	0.0859	0.0868
K(215,0.95,0.95), one sided tolerance factor Ref. D-2)	1.830	1.830
DNBR(L) = 1/(M/P - Kσ) = 1/[1.0040 - 1.830(0.0868)]	1.175	1.183

Parameter Ranges

Pressure, psia	400 to 2465
Mass Velocity, Mlbm/hr-ft^2	0.36 to 3.55
Thermodynamic Quality at CHF	less than 0.74
Thermal-Hydraulic Computer Code	VIPRE-01
Spacer Grid	Mark-B11 15x15 Mixing Vane
Design Limit DNBR, VIPRE-01	1.19*

* The correlation design limit DNBR (1.19) applies only at or above a nominal pressure of 1000 psia (Reference D-2). In the low pressure region (below a nominal pressure of 1000 psia) the design limit DNBR in the following table will be used (Reference D-2):

Pressure	Design Limit DNBR
400 to 700 psia	1.59
700 to 1000 psia	1.20

TABLE D-4 Oconee Statistically Treated Uncertainties

Parameter	Type	Type of Distribution	Uncertainty	Standard Deviation
Reactor System				
Core Power*	Measurement	Normal	+/-2.0%FP	+/-1.0%FP
Core Flow	Measurement	Normal	4 Pump: +/-2.0%	+/-1.0%
			3 Pump: +/-3.2%	+/-1.6%
			2 Pump: +/-4.2% design	+/-2.1% design
Pressure	Measurement	Normal	+/-30.0 psi	+/-15.0 psi
Temperature	Measurement	Normal	+/-2.0°F	+/-1.0°F
Nuclear				
FΔH	Calculation	Normal	---	+/-2.84%
Fz	Calculation	Normal	---	+/-2.91%
Z	Calculation	Uniform	+/-6 inches	---
Fq	Calculation	Normal	[	]
Hot Channel Flow Area	Measurement	Uniform	[ ]	---
DNBR	Correlation	Normal	---	9.268%
DNBR	Code	Normal	[	]

* Percentage of 100% RTP (69.75 MWth wherever applied).

REFERENCES

D-1. DPC-NE-2003P-A, Oconee Nuclear Station Core Thermal-Hydraulic Methodology Using VIPRE-01, October 1989.

D-2. The BWU Critical Heat Flux Correlations, Addendum 1 to BAW-10199P-A, Framatome Cogema Fuels, Lynchburg, Virginia, April 6, 2000.

D-3. VIPRE-01: A Thermal-Hydraulic Code For Reactor Cores, EPRI NP-2511-CCM-A, Vol. 1-4, Battelle Pacific Northwest Laboratories, August 1989.

D-4 DPC-NE-2005P-A, Rev. 1, Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology, Appendix C, November 1996.

Appendix E and Response to Request for Additional Information

Revision 3

DPC-NE-2005A

Duke Power Thermal-Hydraulic Statistical Core Design Methodology

APPENDIX E

McGuire/Catawba Plant Specific Data

Advanced Mark-BW, BWU-Z CHF Correlation

Submitted: September 2001
Approved: September 2002

This Appendix contains the plant specific data and statistical DNB limits for the McGuire and Catawba Nuclear Stations with the Advanced Mark-BW fuel design using the BWU-Z critical heat flux correlation. The thermal-hydraulic statistical core design analysis was performed as described in the main body of this report (DPC-NE-2005).

This appendix details the fuel assembly structural and thermal-hydraulic features unique to the Advanced Mark-BW fuel design. Two separate fuel pellet materials can be used in this structure. When used with uranium fuel pellets, the fuel assembly is called Advanced Mark-BW. If used with mixed oxide fuel pellets, the fuel assembly is called Mark-BW/MOX1. The fuel mechanical structure and grids are identical in each case, therefore the same critical heat flux correlation is applicable to both designs. The nuclear uncertainties used in this analysis bound both uranium and mixed oxide fuel rods. Therefore, the SCD analysis documented here is applicable to and bounds both the Advanced Mark-BW and the Mark-BW/MOX1 fuel designs. For simplicity in this appendix, the term Advanced Mark-BW will be used.

Plant Specific Data

This analysis is for the McGuire and Catawba plants (four loop Westinghouse PWR's) with the Advanced Mark-BW fuel. This fuel design incorporates a 17x17 fuel lattice with 0.374 inch outside diameter (OD) fuel rods, M5™ cladding, and three additional non-structural Mid-Span Mixing (MSM) grids in the upper fuel assembly spans to improve DNB performance. All the parameter uncertainties and statepoint ranges used in this analysis were selected to bound the unit and cycle specific system values at the McGuire and Catawba stations.

Thermal Hydraulic Code and Model

The VIPRE-01 thermal-hydraulic computer code described in Reference E-3 and the McGuire/Catawba eight channel model approved in Reference E-1 are used in this analysis. The reference pin power distribution is the same as that used for the Westinghouse RFA fuel described in Reference E-4. The VIPRE-01 models, approved in Reference E-1 for the Mark-BW fuel, are used to analyze the Advanced Mark-BW fuel design with the following changes:

1) The Advanced Mark-BW fuel assembly geometry information is listed in Table E-1. Applicable form loss coefficients as per the vendor were used in the model.
2) The bulk void fraction model was changed from the Zuber-Findlay model to the EPRI model. Correspondingly, the subcooled void model was changed from LEVY to the EPRI model.

The Zuber-Findlay bulk void model is applicable only to qualities below approximately 0.7 (void fractions of 0.85) and is discontinuous at a quality equal to 1.0 (Reference E-3). The EPRI bulk void model is essentially the same as the Zuber-Findlay bulk void model except for the equation used to calculate the drift velocity (Reference E-3). This eliminates the discontinuity at high qualities and void fractions. Therefore, the EPRI model covers the full range (i.e., void fraction range, 0 - 1.0) of void fractions required for performing DNB calculations. Also, for overall void model compatibility, the subcooled void model was changed from the Levy model, as specified in Reference E-1, to the EPRI correlation. This change has been previously submitted and approved by the NRC for both the Westinghouse RFA fuel design (Reference E-4) and the Mark-B11 fuel design (DPC-NE-2005, Revision 2, Appendix D).

Critical Heat Flux Correlation

The BWU-Z critical heat flux correlation described in Reference E-2 is used for all statepoint analyses. This correlation was developed by Framatome Cogema Fuels and is applicable to the Advanced Mark-BW fuel design. The analysis in Reference E-2 was performed with the LYNXT thermal-hydraulic computer code. This correlation was programmed into the VIPRE-01 thermal-hydraulic computer code and the Advanced Mark-BW fuel database was analyzed in its entirety. The results of this analysis are shown in Table E-2. The resulting average measured to predicted (M/P) value and data standard deviation are within 1% and the CHF correlation limit with VIPRE-01 is 2% lower than the values in Reference E-2, page F-5 (also shown on Table E-2 under the LYNXT column).

Figures E-1 through E-4 graphically show the results of this evaluation. Figure E-1 shows there is no bias of measured CHF values to VIPRE-01 predicted values for the database. Figures E-2 through E-4 show there is no bias with the VIPRE-01 calculated M/P ratios with respect to mass velocity, pressure, or thermodynamic quality. These figures compare closely with the same parameter representations in Reference E-2.

Based on the results shown in Table E-2 and Figures E-1 through E-4, the BWU-Z form of the BWU CHF application correlation can be used in DNBR calculations with VIPRE-01 for Advanced Mark-BW fuel.

Statepoints

The statepoint conditions evaluated in this analysis are listed in Table E-3. These statepoints represent the range of conditions to which the statistical DNB analysis limit will be applied.

Key Parameters and Uncertainties

The key parameters and their uncertainty magnitude and associated distribution used in this analysis are listed on Table E-4. The uncertainties were selected to bound the values calculated for each parameter at McGuire and Catawba. As noted in Table E-4, the nuclear uncertainties used in this analyses bound both uranium and mixed oxide fuel. The resulting range of key parameter values generated in this analysis is listed on Table E-6.

Mixed Core Application

The mixed core model determines the impact of the geometric and hydraulic differences between the resident 17x17 Westinghouse RFA fuel described in Reference E-4 and the new Advanced Mark-BW design. The 8 channel model described in Reference E-1 is used to evaluate the impact of mixed cores containing Westinghouse RFA fuel and the Advanced Mark-BW fuel. In Figure 5 of Reference E-1, Advanced Mark-BW fuel is used instead of Mark-BW fuel. Therefore, the limiting assembly in Channels 1 through 7 are modeled as Advanced Mark-BW fuel and the remaining core, Channel 8, is modeled as Westinghouse RFA fuel. The mixed core analysis models each fuel type in those respective locations with the correct geometry. The form loss coefficients for each fuel design are input so the effect of crossflow between the different fuel types by elevation is calculated. This conservative mixed core model is used for all analyses since the equilibrium core reload cycles will contain both fuel types.

DNB Statistical Design Limits

The statistical design limit for each statepoint evaluated is listed on Table E-5. Section 1 of Table E-5 contains the 500 case runs and Section 2 contains the 6000 case runs. The number of cases was increased from 5000 to 6000 as described in Attachment 1 of Revision 0 of DPC-NE-2005. The DNBR distributions for all statepoints in this analysis were normally distributed. It is seen from Section 2 of Table E-5 that the maximum statepoint statistical DNBR value is [ ]. Therefore, the statistical design limit using the BWU-Z CHF correlation for Advanced Mark-BW fuel at McGuire/Catawba was conservatively determined to be 1.36. This limit applies to mixed cores with Advanced Mark-BW and Westinghouse RFA fuel.

FIGURE E-1
Measured CHF versus Predicted CHF
Advanced Mark-BW Fuel Database

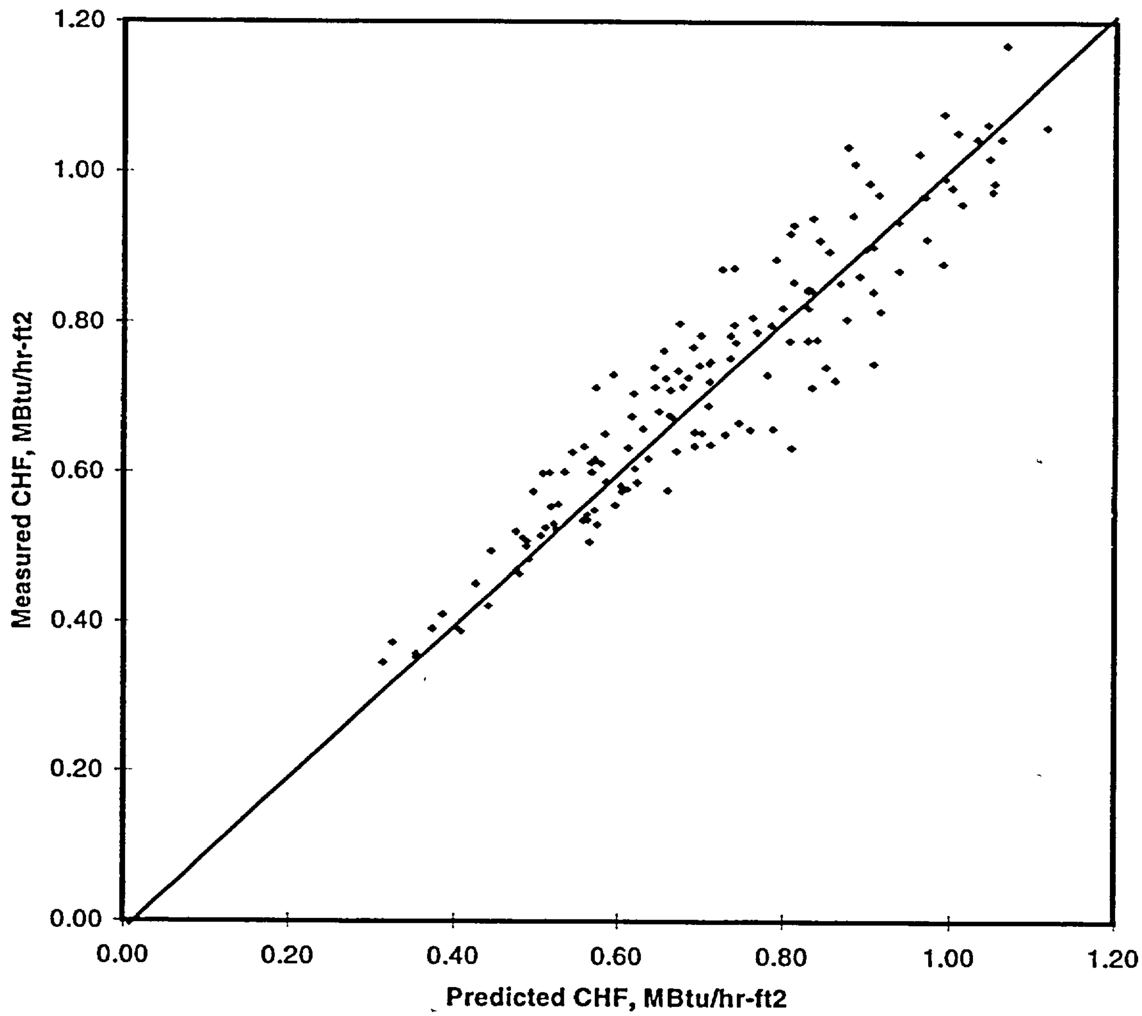

FIGURE E-2
Measured to Predicted CHF versus Mass Velocity
Advanced Mark-BW Fuel Data Base

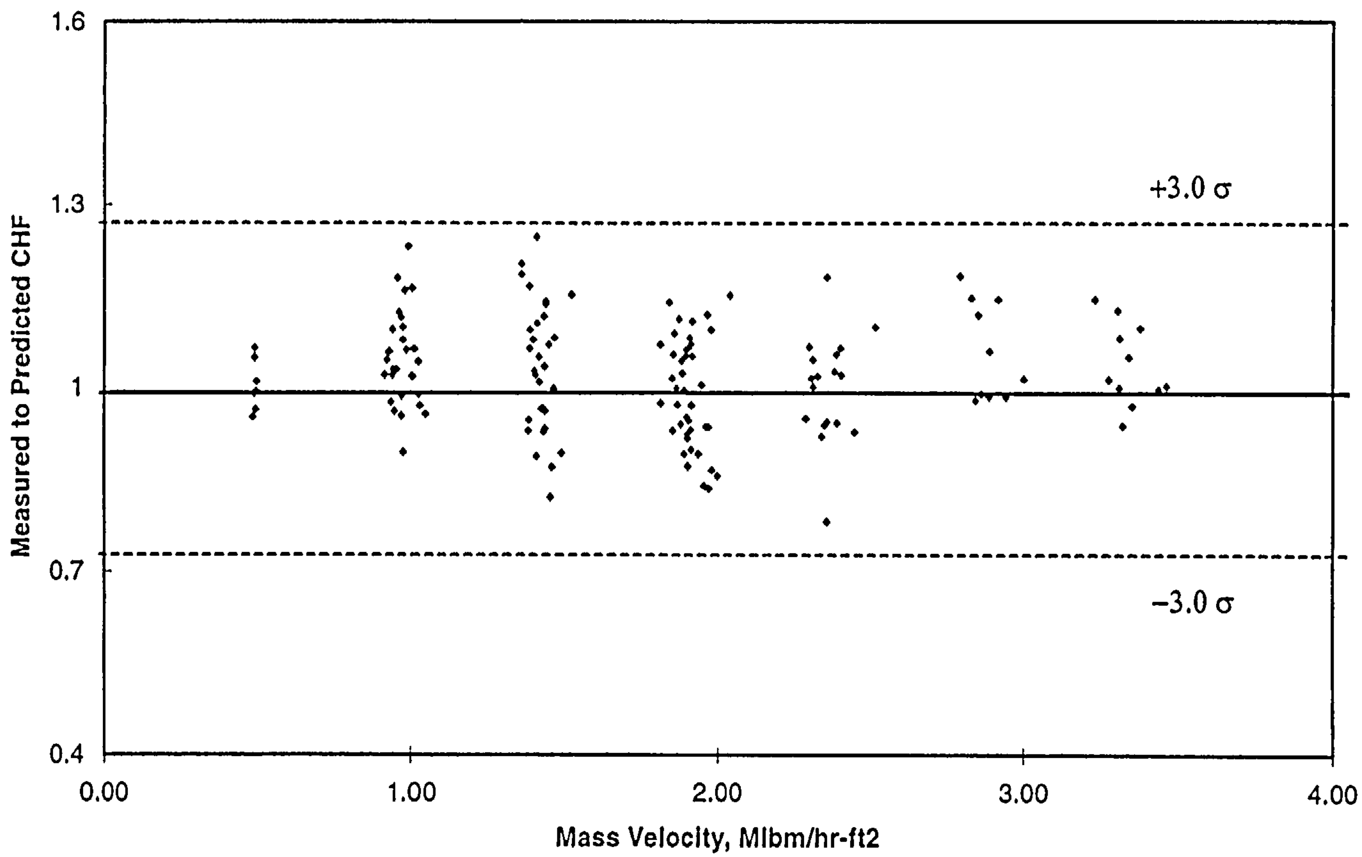

FIGURE E-3
Measured to Predicted CHF versus Pressure
Advanced Mark-BW Fuel Data Base

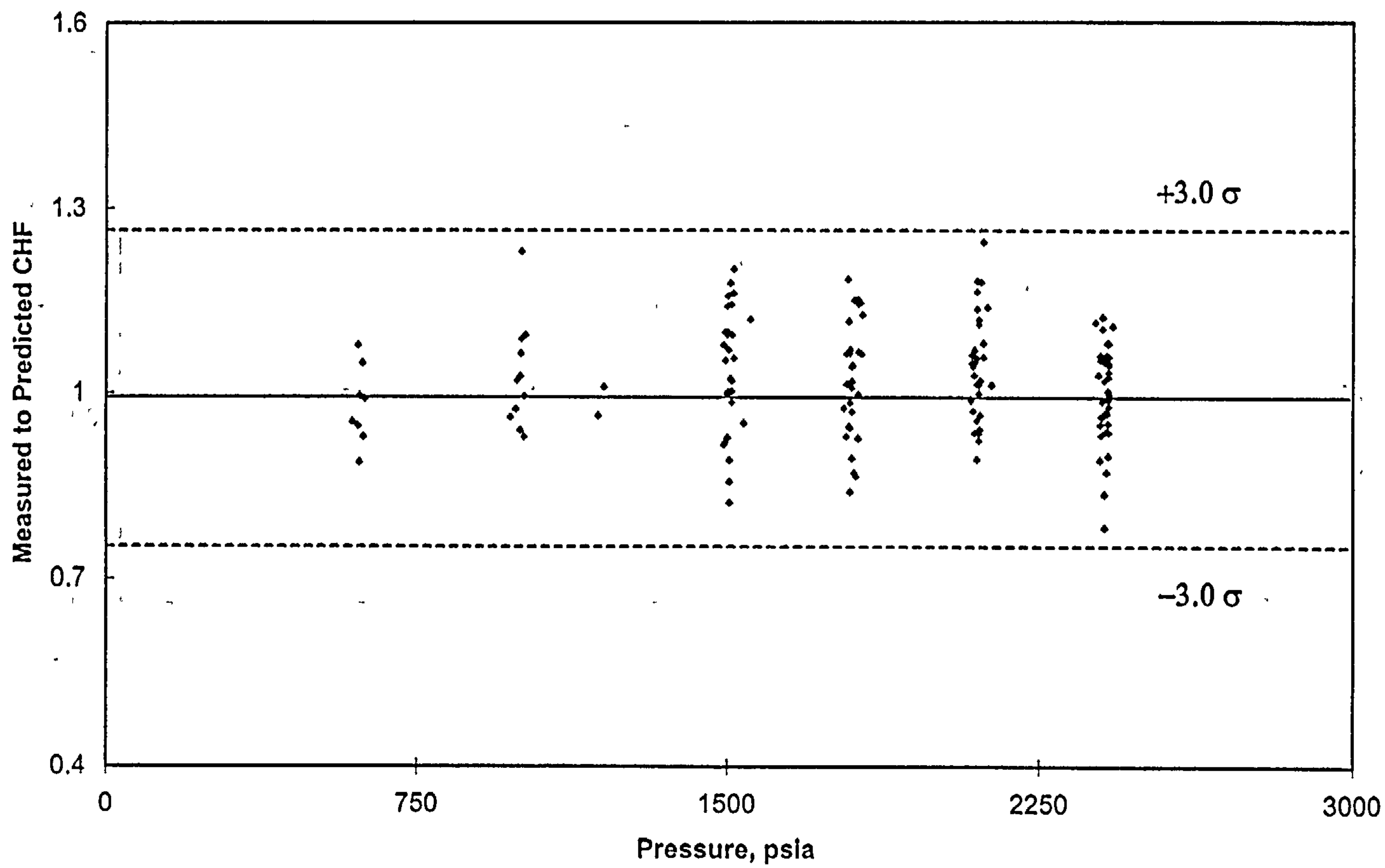

FIGURE E-4
Measured to Predicted CHF versus Quality
Advanced Mark-BW Fuel Data Base

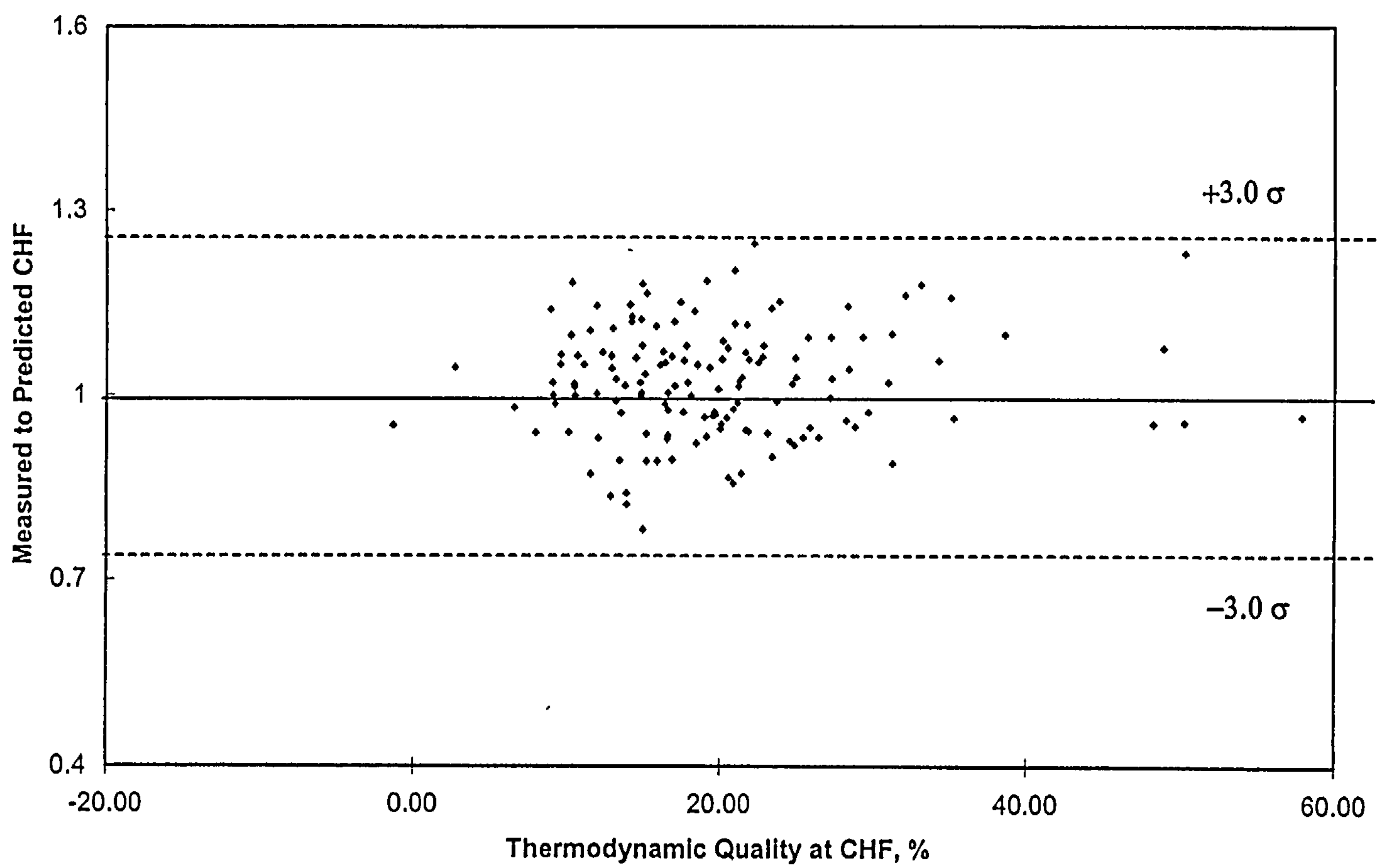

TABLE E-1 Advanced Mark-BW Fuel Assembly Data

(TYPICAL)

GENERAL FUEL SPECIFICATIONS

Fuel rod diameter, inches (Nominal)	0.374
Guide tube diameter, inches (Nominal)	0.482
Fuel rod pitch, inches (Nominal)	0.496
Fuel Assembly pitch, inches (Nominal)	8.466
Fuel Assembly length, inches (Nominal)	160.0

GENERAL FUEL CHARACTERISTICS

Component	Material	Number	Location	Type
Grids	Inconel	2	Upper and Lower	Non-Mixing Vane
	Zircaloy (M5™)	6	Intermediate	5 Vaned, 1 Vaneless
	Zircaloy (M5™)	3	Intermediate	Mid-Span Mixing (Non-structural)
Nozzles	304SS	1	Bottom	Fine Mesh
	304SS	1	Top	Quick Disconnect

TABLE E-2 CHF Test Database Analysis Results With VIPRE-01 Advanced Mark-BW Fuel, BWU-Z CHF Correlation

VIPRE-01/LYNXT Statistical Results

	VIPRE-01	LYNXT
n, # Of data	148	148
N, degrees of freedom (n-1)	147	147
M/P, Average measured to predicted CHF	1.0214	1.0138
σ (M/P/N)	0.0883	0.0920
K(147,0.95,0.95),one sided tolerance factor Ref. E-2)	1.872	1.872
DNBR(L) = 1 /(M/P – K * σ)	1.168	1.188

BWU-Z Parameter Ranges

Pressure, psia	400 to 2465
Mass Velocity, Mlbm/hr-ft^2	0.36 to 3.55
Thermodynamic Quality at CHF	Less than 0.74
Thermal-Hydraulic Computer Code	VIPRE-01
Spacer Grid	Advanced Mark-BW, F_{MSM} = 1.18
Design Limit DNBR, VIPRE-01	1.19*

* The correlation design limit DNBR (1.19) applies only at or above the nominal pressure of 1000 psia (Reference E-2). In the low pressure region (below a pressure of 1000 psia) the design limit DNBR in the following table will be used (Reference E-2):

Pressure	Design Limit DNBR
400 to 700 psia	1.59
700 to 1000 psia	1.20

TABLE E-3 McGuire/Catawba SCD Statepoints

Stpt No.	Power* (% RTP)	RCS Flow (K gpm)	Pressure (psia)	Core Inlet Temperature (°F)	Axial Peak (F_z @ Z)	Radial Peak# (FΔH)
1						
2						
3						
4						
5						
6						
7						
8						
9						
10						
11						
12						
13						
14						
15						
16						
17						
18						
19						
20						
21						
22						
23						
24						
25						

* 100% RTP = 3411 Megawatts Thermal

\# FΔH is maximum pin peak

TABLE E-4 McGuire/Catawba Statistically Treated Uncertainties

Parameter	Uncertainty / Standard Deviation	Type of Distribution
Core Power*	+/- 2% / +/- 1.22%	Normal
Core Flow		
Measurement	+/- 2.2% / +/- 1.34%	Normal
Bypass Flow	+/- 1.5%	Uniform
Pressure	+/- 30 psi	Uniform
Temperature	+/- 4 deg F	Uniform
$F^N_{\Delta H}$		
Measurement	+/- 4.0% / 2.43%	Normal
$F^E_{\Delta H}$	+/- 3.5% / 2.13%	Normal
Spacing	+/- 2.0% / 1.22%	Normal
F_Z	+/- 4.4% / 2.68%	Normal
Z	+/- 6 inches	Uniform
DNBR		
Correlation	15.3% / +/- 9.27%	Normal
Code/Model	[ ]	Normal

* - Percentage of 100% RTP (68.22 MWth wherever applied).

TABLE E-4 Continued McGuire/Catawba Statistically Treated Uncertainties

Parameter	Justification
Core Power	The core power uncertainty was calculated by statistically combining the uncertainties of the process indication and control channels. The uncertainty is calculated from normally distributed random error terms such as sensor calibration accuracy, rack drift, sensor drift, etc. combined by the square root sum of squares method (SRSS). Since the uncertainty is calculated from normally distributed values, the parameter distribution is also normal.
Core Flow	
Measurement	Same approach as Core Power.
Bypass Flow	The core bypass flow is the parallel core flow paths in the reactor vessel (guide thimble cooling flow, head cooling flow, fuel assembly/baffle gap leakage, and hot leg outlet nozzle gap leakage) and is dependent on the driving pressure drop. Parameterizations of the key factors that control ΔP, dimensions, loss coefficient correlations, and the effect of the uncertainty in the driving ΔP on the flow rate in each flow path, was performed. The dimensional tolerance changes were combined with the SRSS method and the loss coefficient and driving ΔP uncertainties were conservatively added to obtain the combined uncertainty. This uncertainty was conservatively applied with a uniform distribution.
Pressure	The pressure uncertainty was calculated by statistically combining the uncertainties of the process indication and control channels. The uncertainty is calculated from random error terms such as sensor calibration accuracy, rack drift, sensor drift, etc. combined by the square root sum of squares method. The uncertainty distribution was conservatively applied as uniform.
Temperature	Same approach as Pressure.

TABLE E-4 Continued McGuire/Catawba Statistically Treated Uncertainties

Parameter	Justification
$F^N_{\Delta H}$	
Measurement	This uncertainty is the measurement uncertainty for the movable incore instruments. A measurement uncertainty can arise from instrumentation drift or reproducibility error, integration and location error, error associated with the burnup history of the core, and the error associated with the conversion of instrument readings to rod power. The uncertainty distribution is normal. This uncertainty is bounding for both uranium and mixed oxide fuel.
$F^E_{\Delta H}$	This uncertainty accounts for the manufacturing variations in the variables affecting the heat generation rate along the flow channel. This conservatively accounts for possible variations in the pellet diameter, density, and U_{235} enrichment. This uncertainty distribution is normal and was conservatively applied as one-sided in the analysis to ensure the MDNBR channel location was consistent for all cases. This uncertainty bounds both uranium and mixed oxide fuel pellets.
Spacing	This uncertainty accounts for the effect on peaking of reduced hot channel flow area and spacing between assemblies. The power peaking gradient becomes steeper across the assembly due to reduced flow area and spacing. This uncertainty distribution is normal and was conservatively applied as one-sided to ensure consistent MDNBR channel location.
F_Z	This uncertainty accounts for the axial peak prediction uncertainty of the physics codes. The uncertainty distribution is applied as normal.
Z	This uncertainty accounts for the possible error in interpolating on axial peak location in the maneuvering analysis. The uncertainty is one physics code axial node length. The uncertainty distribution is conservatively applied as uniform.

TABLE E-4 Continued McGuire/Catawba Statistically Treated Uncertainties

Parameter	Justification
DNBR	
Correlation	This uncertainty accounts for the CHF correlation's ability to predict DNB. The LYNXT value was used since the VIPRE-01 value was smaller. The uncertainty distribution is applied as normal.
Code/Model	This uncertainty accounts for the thermal-hydraulic code uncertainties and offsetting conservatisms. This uncertainty also accounts for the small DNB prediction differences between the various model sizes. The uncertainty distribution is applied as normal.

TABLE E-5 McGuire/Catawba Statepoint Statistical Results

SECTION 1

BWU-Z Critical Heat Flux Correlation

500 Case Runs

Statepoint #	Mean	σ	Coefficient of Variation	Statistical DNBR
1				
2				
3				
4				
5				
6				
7				
8				
9				
10				
11				
12				
13				
14				
15				
16				
17				
18				
19				
20				
21				
22				
23				
24				
25				

TABLE E-5 Continued McGuire/Catawba Statepoint Statistical Results

SECTION 2

BWU-Z Critical Heat Flux Correlation

6000 Case Runs

Statepoint #	Mean	σ	Coefficient of Variation	Statistical DNBR
1				
6				
7				
10				
11				
12				

TABLE E-6 McGuire/Catawba Key Parameter Ranges

Parameter	Maximum	Minimum
Core Power* (% RTP)		
Pressure (psia)		
T inlet (deg. F)		
RCS Flow (Thousand GPM)		
FΔH, Fz, Z		

* 100% RTP = 3411 Megawatts Thermal

All values listed in this table are based on the currently analyzed Statepoints. Ranges are subject to change based on future statepoint conditions.

REFERENCES

E-1. DPC-NE-2004P-A, McGuire and Catawba Nuclear Station Core Thermal-Hydraulic Methodology Using VIPRE-01, Revision 1, February 1997.

E-2. BAW-10199P-A, Addendum 2, Application of the BWU-Z CHF Correlation to the Mark-BW17 Fuel Design with Mid-Span Mixing Grids, March 2002.

E-3. VIPRE-01: A Thermal-Hydraulic Code For Reactor Cores, EPRI NP-2511-CCM-A, Vol. 1-4, Battelle Pacific Northwest Laboratories, August 1989.

E-4. DPC-NE-2009P-A, Duke Power Company Westinghouse Fuel Transition Report, December 1999.

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August 14, 2002

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555-0001

Subject: McGuire and Catawba Nuclear Station, Units 1 and 2
Docket Nos. 50-369, 50-370, 50-413, 50-414
Topical Report DPC-NE-2005P, *Thermal-Hydraulic Statistical Core Design Methodology*, Revision 3 (Appendix E); Request for Additional Information

Reference: U.S. Nuclear Regulatory Commission letter dated August 1, 2002, Request for Additional Information - Review of Duke Topical Report DPC-NE-2005P, (TAC Nos. MB3105, MB3106, MB3173 and MB3175)

Duke Power Company's hereby submits its response to the Nuclear Regulatory Commission (NRC) Request for Additional Information (RAI) transmitted by the reference letter. This submittal consists of two attachments; Attachment 1 is a proprietary version of Duke's response to the RAI and Attachment 2 is a non-proprietary version. The proprietary information in Attachment 1 is enclosed in brackets [ ]. In accordance with 10 CFR 2.790, Duke requests that this proprietary information be withheld from public disclosure. An affidavit that attests to the proprietary nature of this information is included with this letter.

Also included in this submittal is a revised References page (E-20) for DPC-NE-2005P reflecting NRC approval of Framatome ANP Topical Report BAW-10199P-A, Addendum 2.

Inquiries on this matter should be directed to G.A Copp at (704) 373-5620.

Very truly yours,

K. S. Canady

Attachments

xc w/attachments:

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AFFIDAVIT OF K.S. CANADY

1. My name is K. S. Canady. I am Vice President of Duke Energy Corporation, and as such have the responsibility of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear plant licensing, am authorized to apply for its withholding on behalf of Duke.

2. I am making this affidavit in conformance with the provisions of 10 CFR 2.790 of the regulations of the Nuclear Regulatory Commission (NRC) and in conjunction with Duke's application for withholding which accompanies this affidavit.

3. I have knowledge of the criteria used by Duke in designating information as proprietary or confidential.

4. Pursuant to the provisions of paragraph (b)(4) of 10 CFR 2.790, the following is furnished for consideration by the NRC in determining whether the information sought to be withheld from public disclosure should be withheld.

   (i) The information sought to be withheld from public disclosure is owned by Duke and has been held in confidence by Duke and its consultants.

   (ii) The information is of a type that would customarily be held in confidence by Duke. The information consists of analysis methodology details, analysis results, supporting data, and aspects of development programs, relative to a method of analysis that provides a competitive advantage to Duke.

   (iii) The information was transmitted to the NRC in confidence and under the provisions of 10 CFR 2.790, it is to be received in confidence by the NRC.

   (iv) The information sought to be protected is not available in public to the best of our knowledge and belief.

   (v) The proprietary information sought to be withheld in this submittal is that which is marked in the attachment to Duke Energy Corporation letter dated August 15, 2002; Response to Request for Additional Information Re: Topical Report DPC-NE-2005P, Revision 3, Duke Power Thermal-Hydraulic Statistical Core Design (SCD) Methodology. This information enables Duke to:

      (a) Respond to Generic Letter 83-11, "Licensee Qualification for Performing Safety Analyses in Support of Licensing Actions."

K. S. Canady

(continued)

(b) Support license amendment and Technical Specification revision requests for Duke reactors to support the use of Advanced Mk-BW or Mk-BW/MOX1 fuel assemblies.

(c) Perform safety reviews per 10 CFR 50.59.

(d) Evaluate core thermal-hydraulic performance and support the establishment of core operating limits.

(vi) The proprietary information sought to be withheld from public disclosure has substantial commercial value to Duke.

(a) It allows Duke to reduce vendor and consultant expenses associated with supporting the operation and licensing of nuclear power plants.

(b) Duke intends to sell the information to nuclear utilities, vendors, and consultants for the purpose of supporting the operation and licensing of nuclear power plants.

(c) The subject information could only be duplicated by competitors at similar expense to that incurred by Duke.

5. Public disclosure of this information is likely to cause harm to Duke because it would allow competitors in the nuclear industry to benefit from the results of a significant development program without requiring a commensurate expense or allowing Duke to recoup a portion of its expenditures or benefit from the sale of the information.

K. S. Canady, being duly sworn, on his oath deposes and says that he is the person who subscribed his name to the foregoing statement, and that the matters and facts set forth in the statement are true.

K. S. Canady

Sworn to and subscribed before me this 14TH day of August, 2002.
Witness my hand and official seal.

Mary P. Nehus
Notary Public

My commission expires: JAN 22, 2006

SEAL

Attachment 2
Duke Response to NRC Request for
Additional Information on DPC-NE-2005, Revision 3 (Appendix E)

The staff has reviewed Duke Power's submittal dated September 13, 2001, "Appendix E to DPC-NE-2005P, Duke Power Thermal-Hydraulic Statistical Core Design (SCD) Methodology (Proprietary)" and has identified a need for the following additional information.

1. The submittal states that Appendix E contains the plant specific data and statistical departure from nucleate boiling (DNB) limits for the McGuire and Catawba Nuclear Stations with the Advanced Mark-BW fuel design using the BWU-Z critical heat flux (CHF) correlation and provides the fuel assembly structural and thermal-hydraulic features unique to the Advanced Mark-BW fuel design. However, the submittal also states that its SCD analysis is applicable to and bounds both the Advanced Mark-BW and the Mark-BW/MOX1 fuel designs. It appears that the data provided in the submittal are only applicable to the Advanced Mark-BW fuel design. Please clarify whether the methodology described in Appendix E to DPC-NE-2005P will be applied to both the Advanced Mark-BW and the Mark-BW/MOX1 fuel designs. If it is applicable to both designs, then additional data sets for the Mark-BW/MOX1 should be provided. Also, please identify those differences between the Advanced Mark-BW and Mark-BW17 fuel design.

Response:

To clarify the terminology, a separate description of all three fuel types listed in the question is provided below.

Mark-BW17 Fuel
The Mark-BW17 fuel design is a 17x17 fuel assembly design for operation in Westinghouse NSSS systems by Framatome ANP. The Mark-BW17 fuel assembly was introduced in 1991 and is currently operating in the McGuire, Catawba, and Sequoyah units. The Mark-BW17 includes the following features:

- 0.374 fuel rod outer diameter
- Debris resistant bottom nozzle
- Zircaloy mixing vane spacer grids
- Removable top nozzle

Advanced Mark-BW Fuel
The Advanced Mark-BW fuel design is an evolutionary improvement on the successful Mark-BW17 fuel assembly design. The only thermal-hydraulic difference between the Mark-BW17 fuel and the Advanced Mark-BW fuel is the addition of three mid-span mixing grids to the Advanced Mark-BW design. All other features listed above are the same between Mark-BW17 and the Advanced Mark-BW. Therefore, the Advanced Mark-BW design is the Mark-BW17 fuel with mid-span mixing grids.

Attachment 2
Duke Response to NRC Request for
Additional Information on DPC-NE-2005, Revision 3 (Appendix E)

Mark-BW/MOX1 Fuel Design

The Mark-BW/MOX1 design is virtually identical to the Advanced Mark-BW. The only difference between the designs other than the title is in the fuel rod design. There are two specific differences, namely:

1) The Mark-BW/MOX1 fuel design contains mixed oxide fuel pellets and Advanced Mark-BW fuel design contains UO_2 fuel pellets.
2) The Mark-BW/MOX1 overall fuel rod length is 152.40 inches, 0.24 inches longer than the Advanced Mark-BW fuel rod length of 152.16 inches.

All other fuel assembly hardware, features, and dimensions are the same. The extra fuel rod length described in Item 2 is at the top of the fuel assembly above the end of the heated length and does not impact the DNBR calculations. Therefore, from a thermal-hydraulic fuel feature perspective, there is no difference between the Advanced Mark-BW and the Mark-B/MOX1 fuel designs.

Since the thermal-hydraulic features are the same, the only impact the different fuel rod designs could have on the statistical DNBR limit is the radial and axial nuclear uncertainties. These uncertainties are listed under the headings $F^N_{\Delta H}$ and F_Z in Table E-4 of the submittal. The uncertainty values listed on Table E-4 and used in the analysis bound both mixed oxide and UO_2 fuel rods. Since a larger uncertainty is conservative for the SCD analyses and since both UO_2 and mixed oxide uncertainties are bounded, the methodology and the calculated SCD limit of 1.36 is equally valid for the Advanced Mark-BW and the Mark-BW/MOX1 fuel designs.

As stated in the last sentence of paragraph 2 on page E-1, the term Advanced Mark-BW was used throughout the report for simplicity. The term Advanced Mark-BW in the report means both the Mark-BW/MOX1 and the Advanced Mark-BW design.

2. Provide the Advanced Mark-BW fuel database used in Table E-2 of the September 13, 2001, submittal and describe the process used to obtain the 148 new data points in the two tests. Also, please demonstrate that the new data are duplicate and close to the old data used in the topical report, BAW-10199P, Addendum 2, "Application of the BWU-Z CHF Correlation to the Mark-BW17 Fuel Design with Mid-Span Mixing Grids", and justify that the data base is sufficient for this application.

Response:

The 148 data points analyzed by Duke Energy with the BWU-Z CHF correlation in Appendix E of DPC-NE-2003 are exactly the same 148 data points analyzed in BAW-10199P-A, Addendum 2 by Framatome ANP. Duke Energy analyzed the exact same data base with the VIPRE-01 thermal-hydraulic code. The resulting data in Table E-2 of the report shows how the correlation statistics with VIPRE-01 compared to the LYNXT analysis of the data shown in BAW-10199P-A. The VIPRE-01 code results had a smaller standard deviation compared to LYNXT for the same database. The SCD calculation in DPC-NE-2005, Appendix E, conservatively bounded the larger LYNXT value for the correlation uncertainty.

3. Provide details of the calculation procedure used to evaluate the effect of crossflow between the different fuel types as well as the form loss coefficients used as inputs for the mixed core analysis. Also, describe the real test data available for this application to McGuire and Catawba and justify that the DNB statistical design limit of 1.36 is sufficient for McGuire and Catawba using the BWU-Z CHF correlation for Advanced Mark-BW fuel mixed with Westinghouse robust fuel assembly fuel.

Response:

The procedure used to account for the effects of crossflow between different fuel types in the analysis was to use a conservative mixed core model for all calculations. As stated in Mixed Core Application on page E-4 of the submittal, the 8 channel model was used. The center hot or highest powered assembly, consisting of Channels 1 through 7, is modeled as an Advanced Mark-BW. The remaining core, 192 fuel assemblies, is modeled as Westinghouse RFA fuel in Channel 8. The mixed core analysis models each fuel type in those respective locations with the correct geometry and form losses. The forms loss coefficients by fuel design from top to bottom are as follows:

COMPONENT	ADVANCED MARK-BW	WESTINGHOUSE RFA
Top Nozzle	[	]
Top Inconel		
Mixing Vane Zirc Grids		
IFM / MSMG Grid		
Non-Mixing Vane Zirc Grid		
Bottom Inconel		
Bottom Nozzle	[	]

(1) Inconel grid plus Protective grid

The process for evaluating crossflow is to model the core conservatively with the correct geometry and design data and allow VIPRE-01 to calculate the crossflow in each axial node based on the data. In this manner, the core is accurately modeled and the results represent a conservative mixed core.

With respect to real test data available for this application to McGuire and Catawba, the test data is just as applicable to mixed core configuration as a full core. The Framatome ANP CHF correlation form (BWU) is composed of three parts:

1) a uniform part dependent solely on the local thermal-hydraulic conditions of pressure, mass velocity and thermodynamic quality at the axial location of CHF,
2) a non-uniform F factor modification dependent on the shape of the axial heat flux input, and

3) a multiplicative geometric factor dependent on the overall fuel assembly grid spacing and heated length.

It is with Item 1 (the local conditions thermal-hydraulic conditions) that the mixed core conditions question surfaces. The other two items are unaffected by mixed cores.

CHF correlations are developed from data from full length electrically heated bundles in 5-by-5 rod arrays. For each data point, the inlet conditions of coolant mass velocity, pressure and temperature are known, as is the power (heat flux) required to produce a DNB event. The local thermal-hydraulic conditions at the axial location of CHF must then be calculated with a computer code.

The proof of applicability of a CHF correlation, then, is how well it can predict the critical heat flux that was measured in the DNB event using the calculated local conditions. Thus, the applicability of a CHF correlation is dependent not only on its form and data base, but on the accuracy with which the thermal-hydraulic code calculates local conditions. Because of the size of the test section (a 5-by-5 rod array) and the use of the same spacer grids and elevations, normal CHF tests do not exhibit large hydraulic differences. Framatome ANP, however, has performed one test with widely varying subchannel resistances producing the large velocity upsets representative of mixed core conditions. This test was a 5-by-5 test of the Mark B zircaloy grid modeled as the corner intersection of four fuel assemblies. Laser Doppler Velocimetry (LDV) testing of the intersection grid showed velocity depressions as large as 50% between the intersection subchannel and the surrounding unit cell subchannels. This CHF test was conducted at the Babcock & Wilcox Alliance Research Center and is documented in BAW-10143P-A (BWC correlation of Critical Heat Flux, April, 1985). In the topical, the measured to predicted (M/P) CHF results were compared for two traditional test bundles and the intersection bundle. The guide tube bundle (B15) had an average M/P of 0.971, the unit cell bundle (B16) 0.985 and the intersection bundle (B17) 0.976. The difference in M/P results is statistically insignificant.

The predicted local conditions for the unit and guide tube bundles had very little hydraulic upset, while the intersection bundle (conditions representative of a mixed core) had severe predicted upsets, similar to the measured data. The fact that the CHF correlation performed consistently on conditions representative of both homogeneous and mixed cores confirms that the thermal-hydraulic codes predict the right conditions even with large velocity upsets. This confirms the Framatome ANP CHF correlations are valid for both homogeneous and mixed core applications.